POLICY ISSUE
NOTATION VOTE

July 7, 2011 SECY-11-0089

FOR: The Commissioners

FROM: R. W. Borchardt
Executive Director for Operations

SUBJECT: OPTIONS FOR PROCEEDING WITH FUTURE LEVEL 3 PROBABILISTIC RISK ASSESSMENT ACTIVITIES

PURPOSE:

In response to staff requirements memorandum (SRM) M100218 (ML100780578), this paper (1) provides the Commission with potential future uses for Level 3 probabilistic risk assessments (PRAs) for nuclear power plants (NPPs), (2) provides the Commission with three primary options for proceeding with future Level 3 PRA activities[1] including resource estimates, (3) informs the Commission of the internal coordination efforts and external stakeholder engagement activities in which the staff participated to formulate its plan and scope for future Level 3 PRA activities, and (4) seeks Commission approval for the staff's recommendation to proceed with focused research to address identified gaps in existing PRA technology before performing a full-scope comprehensive site Level 3 PRA for an operating NPP[2].

SUMMARY:

During a February 2010 Commission meeting on research programs, the staff proposed a scoping study that would evaluate the feasibility of performing a new full-scope comprehensive site Level 3 PRA. This proposal was based on technical advances since the last U.S. Nuclear

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[1] This SECY paper and its enclosures distinguish between "Level 3 PRA activities" and "Level 3 PRAs." The latter refers to a PRA that includes specific technical elements or analyses to assess the public risk from an NPP, while the former refers to activities (e.g., research and development) specifically related to or in support of Level 3 PRAs.

[2] As used in this SECY paper and its enclosures, a full-scope comprehensive site Level 3 PRA is a PRA that includes a quantitative assessment of the public risk from accidents involving all site reactor cores and spent nuclear fuel that can occur during any plant operating state, and that are caused by all initiating event hazards (internal events, fires, flooding, seismic events, and other site-specific external hazards).

Regulatory Commission (NRC)-sponsored Level 3 PRAs were performed in the late 1980s and an interest in enhancing the scope of previous and current PRAs to include an assessment of the risk from accidents involving additional site radiological sources (e.g., spent nuclear fuel and multiple units[3]). This SECY paper summarizes the staff's response to SRM M100218 in which the Commission expressed conditional support for Level 3 PRA related activities and directed the staff to provide various options for proceeding with Level 3 PRAs. This paper also summarizes the staff's approach including scoping study objectives and internal coordination and external stakeholder engagement activities. In addition, this paper discusses potential future uses for Level 3 PRAs and presents three primary options for proceeding with future Level 3 PRA activities, including resource estimates. Finally, based on challenges created by the existing budget climate and the additional estimated resources needed, this SECY paper provides the staff's recommendation to proceed with conducting focused research to address identified gaps in existing PRA technology before performing a full-scope comprehensive site Level 3 PRA for an operating NPP.

This SECY paper includes two enclosures. The first enclosure provides more detailed information on (1) the basis for originally proposing a scoping study to evaluate the feasibility of performing a new full-scope comprehensive site Level 3 PRA for an NPP, (2) potential future uses for Level 3 PRAs, (3) the three primary options for proceeding with future Level 3 PRA activities, and (4) the activities that supported development of items 2 and 3. The second enclosure provides more detailed information on the structure and evolution of PRA and risk-informed regulation that led to the staff's original proposal for a scoping study to evaluate the feasibility of performing a new full-scope comprehensive site Level 3 PRA.

BACKGROUND:

In 1995, the Commission established the current framework for risk-informed regulation by issuing a PRA Policy Statement[4] that stated the use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art and in a manner that complements NRC's deterministic approach and traditional defense-in-depth philosophy.

PRA is a structured, analytical process that provides both qualitative insights and quantitative estimates of risk by (1) identifying potential sequences that can challenge system operations and lead to an adverse event, (2) estimating the likelihood of these sequences, and (3) estimating the consequences associated with these sequences, if they were to occur. By prioritizing significant risk contributors[5] and characterizing key sources of uncertainty and their impact on results, PRA serves as a useful decisionmaking tool that can help focus thinking and limited agency resources to ensure safety. Moreover, a full-scope comprehensive site Level 3 PRA that includes an assessment of accidents involving the reactor core as well as accidents involving other site radiological sources (e.g., spent fuel pools [SFPs], dry storage casks, and multiple units) can provide valuable insights into the relative importance of various risk contributors. These insights can be used to enhance regulatory decisionmaking and to help

[3] As used in this SECY paper and its enclosures, a unit refers to a reactor core and, if applicable, an associated spent fuel pool.

[4] 60 FR 42622, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (August 16, 1995).

[5] As used in this SECY paper and its enclosures, risk contributors include: radiological sources (e.g., reactor core, spent nuclear fuel); initiating event hazards (e.g. internal events, fires, flooding, seismic events, other site-specific external hazards); POSs; accident sequences; failure of structures, systems, and components; and operator actions.

focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety.

Using information from Level 3 PRAs performed in the NUREG-1150 study[6], the staff determined that the reactor-specific risk metrics core damage frequency (CDF) and large early release frequency (LERF) can be used respectively as surrogates for the latent cancer risk and prompt fatality risk quantitative health objectives defined in the Commission's Safety Goal Policy Statement[7]. Therefore, instead of using Level 3 PRAs, the staff compares the results from Level 1 and limited-scope Level 2 PRAs to subsidiary numerical objectives based on CDF and LERF for regulatory decisionmaking involving plant-specific applications[8]. Although Level 3 PRAs have since been performed to some extent within both the United States and international nuclear industries, NRC has not sponsored development of a Level 3 PRA since NUREG-1150.

The staff has identified several compelling reasons for proceeding with a new full-scope comprehensive site Level 3 PRA. In the more than two decades that have passed since the NUREG-1150 Level 3 PRAs were performed, numerous technical advances have been made that were not reflected in the NUREG-1150 PRA models. Examples of such technical advances include (1) modifications to enhance NPP operational performance, safety, and security (e.g., development and implementation of risk-informed regulations; improved operational, maintenance, and training practices; implementation of severe accident management guidelines [SAMGs]; and implementation of extensive damage mitigation guidelines [EDMGs] or B.5.b mitigation strategies[9]); (2) significantly improved understanding and modeling of severe accident phenomena; and (3) advances in PRA technology (e.g., improved methods, models, analytical tools, and data through research and operating experience). The staff has also identified additional scope considerations not previously considered that could be addressed by performing a new full-scope comprehensive site Level 3 PRA. Examples include (1) consideration of multi-unit site effects and (2) consideration of other site radiological sources (e.g., SFPs, dry storage casks, and multiple units). NRC has never sponsored a site Level 3 PRA that includes an assessment of both accidents involving the reactor core of a single unit and accidents involving other site radiological sources. The incorporation of these technical advances and additional scope considerations into a new full-scope comprehensive site Level 3 PRA could yield new and improved risk insights to enhance regulatory decisionmaking and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety.

During the Annual Commission Meeting on Research Programs, Performance, and Future Plans on February 18, 2010, the staff proposed a scoping study to evaluate the feasibility of performing a new full-scope comprehensive site Level 3 PRA. In SRM M100218 dated March 19, 2010, the Commission expressed conditional support for Level 3 PRA related activities and directed the staff to (1) continue internal coordination efforts and engage external stakeholders

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[6] NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" (December 1990).

[7] 51 FR 30028, "Safety Goals for the Operations of Nuclear Power Plants" (August 21, 1986).

[8] Regulatory Guide (RG) 1.174, Rev. 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (May 2011).

[9] EA-02-026, "Order for Interim Safeguards and Security Compensatory Measures" (February 25, 2002). Section B.5.b requires licensees to adopt mitigation strategies using readily available resources to maintain or restore core cooling, containment, and SFP cooling capabilities to cope with the loss of large areas of the facility due to large fires and explosions from any cause, including beyond-design-basis aircraft impacts. These requirements were formalized through rulemaking in 10 CFR 50.54(hh)(2).

in formulating a plan and scope for future actions and (2) provide the Commission with various options for proceeding that include costs and perspectives on future uses for Level 3 PRAs. The remainder of this SECY paper summarizes the staff's response to the SRM and provides the staff's recommendation for proceeding.

DISCUSSION:

APPROACH AND ACTIVITIES

In response to SRM M100218, the staff developed a three-phased approach to planning and conducting future Level 3 PRA activities. The first phase consisted of a scoping study that began in April 2010 and ended upon submission of this SECY paper to the Commission. This scoping study was conducted by staff from the Office of Nuclear Regulatory Research (RES) with support from staff from the Office of Nuclear Material Safety and Safeguards (NMSS), Office of New Reactors (NRO), Office of Nuclear Reactor Regulation (NRR), and Office of Nuclear Security and Incident Response (NSIR). The objectives and activities associated with this scoping study are discussed in more detail below and in Enclosure 1. The second phase would consist of proceeding with either one of the options developed by the staff as part of the scoping study or any other option directed by the Commission following the submission of this SECY paper. Based on the results and insights from the second phase, the staff would then assess the need for follow-on Level 3 PRA activities and then provide the Commission with additional options and recommendations for proceeding.

Based on Commission tasking in SRM M100218, the staff identified the following main objectives for the Level 3 PRA scoping study (1) identify potential future uses for Level 3 PRAs; (2) develop various options for proceeding with future Level 3 PRA activities that include objectives, scope, PRA technology to be used, site selection considerations[10], and resource estimates; (3) determine the feasibility of proceeding with each of the developed options; (4) continue internal coordination efforts to identify the staff's recommendation for proceeding; and (5) engage external stakeholders to obtain their views on the staff's approach, potential future uses for Level 3 PRAs, options for proceeding with future Level 3 PRA activities, and recommendation for proceeding.

Throughout the scoping study, the staff participated in numerous internal coordination and external stakeholder engagement activities. Internal coordination efforts included workshops, coordination and alignment meetings, and internal stakeholder briefings. External stakeholder engagement activities included Advisory Committee on Reactor Safeguards (ACRS) subcommittee and full committee briefings, Regulatory Information Conference presentations, and a Category 2 public meeting with representatives from nuclear industry, vendor, research, interest group, and public media organizations. Overall, stakeholders supported proceeding with a new full-scope comprehensive site Level 3 PRA. Some expressed concern about the resources needed and the agency's ability to complete such a comprehensive study in a reasonable period of time. ACRS proposed a phased approach and schedule that would enable

[10] Because the Commission expressed only conditional support and directed the staff to provide various options for proceeding, the staff did not include selection of a site to participate in a future Level 3 PRA as one of the objectives of the scoping study. Instead, the staff identified various site selection considerations related to the quality and availability of relevant information that could impact the cost of a future Level 3 PRA. These considerations, which are provided in Enclosure 1, can inform future site selection activities if the Commission directs the staff to proceed with a Level 3 PRA.

the staff to complete such a study while minimizing the near-term resource impact[11]. Enclosure 1 and the public meeting summary (ML111400179) provide more detailed information.

POTENTIAL FUTURE USES FOR LEVEL 3 PRAs

In identifying potential future uses for Level 3 PRAs, a logical first step was to identify how the results and risk insights from the NUREG-1150 PRAs were used. In addition, the staff considered potential enhancements that could be made to the use of PRA in the existing risk-informed regulatory framework. In this way, the staff developed a set of potential uses meant to apply to future Level 3 PRAs in general and not specifically to the full-scope comprehensive site Level 3 PRA proposed as Option 3 (the use of which would ultimately depend on its scope and applicability to the larger population of NPP sites). Potential future uses for Level 3 PRAs include (1) confirm the acceptability of the agency's current use of PRA in risk-informed regulatory decisionmaking (e.g., the use of Level 1 and limited-scope Level 2 reactor PRAs to support regulatory applications and the use of RG 1.174 subsidiary numerical objectives based on the reactor-specific risk metrics CDF and LERF); (2) verify or revise regulatory requirements and guidance, particularly those based on NUREG-1150 information (e.g., RG 1.174 and the regulatory analysis guidelines[12] and technical evaluation handbook[13] used by the staff to evaluate the costs and benefits of proposed backfits[14]); (3) support specific risk-informed regulatory applications (e.g., provide the technical basis for risk-informing the regulation of spent fuel storage and handling, siting, and emergency preparedness, and focus the Reactor Oversight Process); (4) develop and pilot test PRA technology, standards, and guidance; (5) prioritize generic safety issues and nuclear safety research programs; (6) develop in-house PRA technical capability and support PRA knowledge management and risk communication activities; and (7) support future risk-informed licensing of new and advanced reactor designs (e.g., resolving issues with small modular reactor (SMR) designs, using risk insights to enhance the safety focus of SMR reviews, and modifying risk-informed regulatory guidance for new reactors)[15]. Enclosure 1 provides more detailed information including specific examples.

OPTIONS FOR PROCEEDING WITH FUTURE LEVEL 3 PRA ACTIVITIES

This section presents summary descriptions of the three primary options deemed by the staff to best frame the choices from a feasibility and cost-benefit perspective and their relative advantages and disadvantages. Enclosure 1 provides more detailed descriptions. Estimated resources for each option are provided below in the "RESOURCES" section.

Option 1: Maintain Status Quo – Continue Evolutionary Development of PRA Technology

This option maintains the status quo in ongoing activities related to the development and implementation of PRA technology and risk-informed regulation. Ongoing and planned research to develop and improve upon existing PRA methods, models, tools, and data would continue on a resource-available basis as driven by program office user need requests (UNRs), Commission tasking, and the agency's long-term research plan (LTRP). As part of its strategic

[11] Letter from Said Abdel-Khalik to The Honorable Gregory B. Jaczko, "Draft SECY Paper, "Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities"" (June 22, 2011).

[12] NUREG/BR-0058, Rev. 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission" (September 2004).

[13] NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook" (January 1997).

[14] Title 10 Section 50.109, of the *Code of Federal Regulations* (10 CFR 50.109). Backfitting.

[15] Although future Level 3 PRAs would not be developed in time to inform the staff's current activities related to these efforts, they could be used to inform related follow-on activities.

LTRP efforts, the staff has identified Level 2 and Level 3 PRA as areas that would benefit from examination of advanced methods, and is performing limited research in these areas. The staff also would continue to monitor relevant developments within both the United States and international nuclear industries.

Advantages
- Is consistent with the current fiscal climate by focusing limited staff and contract support resources on mission-critical work driven by program office UNRs, Commission tasking, and the agency's LTRP.

Disadvantages
- Insights that could be gained by conducting a full-scope comprehensive site Level 3 PRA to enhance regulatory decisionmaking would not be realized.
- Can result in inconsistent and more costly treatment of potential future issues by developing the necessary PRA technology on an ad-hoc basis.

Option 2: Conduct Focused Research to Address Identified Gaps in Existing PRA Technology Before Performing a Full-Scope Comprehensive Site Level 3 PRA
This option involves near-term focused research aimed at addressing identified gaps in existing PRA technology over the next 2 years. The primary objective of this research would be to ensure important technical gaps related to the expanded scope and differing degrees of sophistication in the existing PRA technology used to analyze the risk from various risk contributors are addressed before developing a new full-scope comprehensive site Level 3 PRA. Selection of Option 2 would require separate Commission direction in the future before proceeding with a new full-scope comprehensive site Level 3 PRA.

Examples of gaps in existing PRA technology that could be addressed include modeling of consequential (linked) multiple initiating events; modeling of multi-unit dependencies; post-core damage and external events human reliability analysis (HRA); spent fuel PRA technology; modeling of aqueous transport and dispersion of radioactive materials through surface water, sediments, soils, and groundwater; and Level 2 and Level 3 PRA uncertainty analysis.

Advantages
- Focuses limited available staff and contract support resources on mission-critical work driven by program office UNRs, Commission tasking, and the agency's LTRP.
- Focuses additional staff and contract support resources that have already been requested to support future Level 3 PRA activities on research needed to address identified gaps in existing PRA technology.
- Produces results and insights that would advance the state-of-practice in specific PRA technical elements and thereby enhance NRC's PRA capability in those technical areas.

Disadvantages
- Delays insights that could be gained by conducting a full-scope comprehensive site Level 3 PRA to enhance regulatory decisionmaking.

Option 3: Conduct a Full-Scope Comprehensive Site Level 3 PRA
This option involves planning for and performing a new full-scope comprehensive site Level 3 PRA for an operating NPP. The objectives of this PRA would be to (1) extract new and

improved risk insights to enhance regulatory decisionmaking and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety; (2) enhance PRA capability, expertise, and documentation; and (3) demonstrate the technical feasibility and evaluate the realistic cost of developing new Level 3 PRAs. The scope of this PRA would include (1) site radiological sources—reactor cores, spent fuel pools, and dry storage casks on site; (2) initiating event hazards—internal events, fires, flooding, seismic events, and other site-specific external hazards; (3) plant operating states (POSs)—at-power and low-power/shutdown. The only factors specifically excluded from the scope would be radiological sources involving fresh nuclear fuel and radiological waste, and initiating events involving deliberate malevolent acts (e.g., terrorism and sabotage). Enclosure 1 includes a detailed discussion of PRA technology and site selection considerations.

Research identified in Option 2 also would be conducted as part of this option, but on an accelerated schedule to support the completion of a full-scope comprehensive site Level 3 PRA within 3 years. Option 2 and Option 3 differ only in terms of timing, sequencing, near-term use of resources, and relative advantages and disadvantages.

Advantages
- Provides new and improved risk insights earlier to enhance regulatory decisionmaking and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety.
- Enhances PRA capability, expertise, and documentation earlier to address potential future issues.

Disadvantages
- Is resource-intensive, requiring more staff and contract support resources than currently budgeted.
- Requires reallocation of qualified risk analysts from other ongoing important activities, potentially resulting in delays to reviews of National Fire Protection Association (NFPA) Standard 805[16] license amendments, refinement of Standardized Plant Analysis Risk (SPAR) models, and reviews of PRAs in support of combined license applications.

Other Options

The Commission has considerable flexibility in selecting an option for proceeding with future Level 3 PRA activities. In addition to the three primary options presented above, for example, the staff also considered additional options, such as performing limited-scope Level 3 PRAs to address specific issues, performing full-scope Level 3 PRAs for new or advanced reactor designs (e.g., future SMR designs), and developing Level 3 PRAs based on existing information (e.g., fire PRAs developed to support transition to NFPA 805, State-of-the-Art Reactor Consequence Analysis [SOARCA] project analyses, existing PRAs developed by licensees). In the latter case, missing PRA technical elements could be developed to complete a Level 3 PRA of suitable scope and level of detail. Likewise, similar to Option 2, the staff could conduct specialized projects to facilitate development of a Level 3 PRA by a volunteer licensee. In this case, the staff could obtain the desired risk insights and PRA capability by working closely with licensee personnel in performing the Level 3 PRA. These options, which are essentially derivatives of the three primary options, would allow the staff to move forward with a Level 3

[16] NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants."

PRA sooner without disrupting other high priority work that requires the near-term attention of a limited number of qualified risk analysts. Regardless of which option is selected, the staff will develop a detailed project and resourcing plan to accomplish the Commission's direction.

Where appropriate, the staff plans to use advanced tools such as the MELCOR severe accident analysis code and the MELCOR Accident Consequence Code System Version 2 (MACCS2) that were used in the SOARCA project, to support future Level 3 PRA activities. The staff recognizes the potential benefits of both PRA and SOARCA methods and tools that should be considered within overall agency resources. PRA can provide greater breadth in modeling an NPP site to capture accident sequences of possible importance while the SOARCA methods and tools can provide details about why the sequence is important as well as mitigation options for the more important accident sequences revealed by a PRA model. This SECY paper provides the staff's evaluation of potential benefits and estimated costs for proposed options for proceeding with future Level 3 PRA activities. A separate evaluation of the potential benefits and costs for further SOARCA-type analyses will be submitted to the Commission in a paper at the conclusion of the SOARCA project in fiscal year (FY) 2012.

RECOMMENDATIONS:

Based on the need to otherwise reallocate a limited number of qualified risk analysts from other priority assignments, the staff recommends the Commission approve Option 2. Selecting this option will enable the staff to use additional resources already requested to support future Level 3 PRA activities to continue important progress toward performing a new full-scope comprehensive site Level 3 PRA. Moreover, this will enable the staff and the Commission to better understand the potential needs and implications of pending recommendations from multiple task forces (e.g., the Chairman's task force to develop options for a more holistic risk-informed, performance-based regulatory approach[17] and the near-term task force to conduct methodical and systematic reviews of our current processes and regulations in response to the recent event in Japan[18]) before committing substantial resources to support a new full-scope comprehensive site Level 3 PRA.

ACRS recommends the staff proceed with a modified version of Option 3 by developing a phased approach and longer schedule for a selected site that will simultaneously minimize the resource impact while still achieving the objectives associated with completing a new full-scope comprehensive site Level 3 PRA. The staff believes Option 2 will provide necessary flexibility to address both current issues and identified technical gaps by not constraining the staff to a particular site. In addition, the staff believes that by first addressing the technical gaps, Option 2 will ultimately result in a more efficient use of resources by better enabling the staff to complete a new full-scope comprehensive site Level 3 PRA within a shorter time period, thereby facilitating continuity in project staff and reducing costly turnover.

Over the next 2 years, the staff will continue to work collaboratively with the program offices and task forces to coordinate and optimize efforts related to (1) ongoing and planned mission-critical work, (2) research to address identified gaps in existing PRA technology, (3) the SOARCA project, and (4) any pending task force recommendations and related subsequent actions. In

[17] Memorandum from G.B. Jaczko to R.W. Borchardt, "Assessment of Options for a More Holistic Risk-Informed, Performance-Based Regulatory Approach" (February 11, 2011).

[18] Tasking Memorandum – COMGBJ-11-0002, "NRC Actions Following the Events in Japan" (March 23, 2011).

addition, the staff will continue to take necessary steps toward performing a new full-scope comprehensive site Level 3 PRA by engaging with the nuclear industry to identify and discuss issues related to the extent of industry participation and site selection. At the end of FY 2013, the staff will reassess the agency's progress and readiness for performing a new full-scope comprehensive site Level 3 PRA and will provide the Commission with options and a recommendation for proceeding.

In preparation for the Commission meeting scheduled for July 28, 2011 to discuss options for proceeding with future Level 3 PRA activities, the staff also recommends releasing this SECY paper to the public as soon as possible. Early release will enhance external stakeholder interactions with the Commission during this meeting by providing external stakeholders who plan to participate with more time to review the staff's final proposal and to prepare a response. Likewise, early release will enhance staff interactions with the Commission by providing the staff with more time to engage with external stakeholders to better understand their position prior to the meeting.

RESOURCES:

The President's Budget for FY 2012 requests 2.0 full-time equivalents (FTE) and $500k in contract support for future Level 3 PRA activities.

For Option 1, which involves maintaining the status quo and not proceeding with near-term Level 3 PRA activities, the 2.0 FTE and $500k already requested for future Level 3 PRA activities in FY 2012 would not be needed and therefore could be reallocated to support other higher priority work.

For Option 2, which involves conducting focused research over the next 2 years to address gaps in existing PRA technology before performing a new full-scope comprehensive site Level 3 PRA, the estimated resources needed are 2.0 FTE and $500k per year for FY 2012 and FY 2013—assuming work commences at the start of FY 2012. For FY 2012, these resources have already been requested to support future Level 3 PRA activities; therefore, Option 2 does not require any additional resource commitments. For FY 2013, the projected resources will be requested through the routine Planning, Budgeting, and Performance Management (PBPM) process.

For Option 3, which involves proceeding with a new full-scope comprehensive site Level 3 PRA for an operating NPP, the estimated resources needed are 8.0 FTE and $2,000k per year for FY 2012, FY 2013, and FY2014. This represents an additional resource commitment of 6.0 FTE and $1,500k per year beyond what has already been requested for FY 2012. These estimates are based on an assumed 3-year project plan and on conservative assumptions with regard to the amount and quality of baseline PRA information available to the staff at the start of the project—which depend on the site selected. The estimated resources needed per year could be reduced by (1) extending the project schedule beyond 3 years, and/or (2) selecting a site willing to participate that has more favorable attributes (e.g., an integrated SPAR model that includes internal and external hazards, low-power/shutdown POSs, and/or Level 2 PRA analyses; developed MELCOR input decks; and/or a developed detailed fire PRA to support transition to NFPA 805).

The additional resources needed to support Option 3 in FY 2012 would require reallocation of resources assigned to ongoing and planned mission-critical work. Should the Commission direct the staff to proceed with Option 3, the staff would first engage with industry to select a site and then develop a detailed project plan that would include more detailed and refined resource estimates; this approach is consistent with ACRS's recommendations. The staff would then coordinate with internal stakeholders to identify the impact of reallocating resources to support the project plan. The detailed project plan, refined resource estimates, and resource impact statements would be provided to the Commission in a separate SECY paper for information prior to proceeding. Projected resources needed to support Option 3 for FY 2013 and FY 2014 would be sought through the routine PBPM process.

To help identify the potential impacts of Option 3 or any other option requiring resources greater than those already requested, the staff examined ongoing and planned work that might be delayed or deferred to support such an option. As noted above, actual decisions regarding impacts would require more refined resource estimates based on site selection and a detailed project and resourcing plan. Examples include:

- **Commission-directed work:** evaluation of different HRA models in an effort to propose either a single model for the agency to use or guidance on which model[s] to use in specific circumstances[19]; development of guidance that will ensure the formal utilization of expert judgment is applied consistently in regulatory decisionmaking[20]; and development of guidance to support risk-informing SMR reviews[21].

- **Program office-requested work:** SPAR model development and improvement to support evaluation of external events and low-power shutdown risk, new reactor designs, and licensee transition to NFPA 805 implementation; simulator research and data collection efforts to improve HRA; development of guidance to support oversight of fitness-for-duty regulations; development of risk-informed methods applicable to security regulation; and support for the extended storage and transportation of spent nuclear fuel initiative.

- **Program office licensing work:** reallocation of 1 FTE and $400k from operating reactor licensing would result in both a per-year reduction of approximately 25 licensing actions or other licensing tasks not related to power uprates or license renewals as well as a delay in the completion of certain licensing actions, including the review of NFPA 805 submittals.

The table below summarizes both the resources already requested to support future Level 3 PRA activities for FY 2012 and the total additional resources needed beyond those already requested for each option for FY 2012, FY 2013, and FY 2014.

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[19] SRM-M061020, "Staff Requirements – Meeting with Advisory Committee on Reactor Safeguards, 2:30 p.m., Friday, October 20, 2006, Commissioners' Conference Room, One White Flint North, Rockville, Maryland (Open to Public Attendance)" (November 8, 2006).

[20] SRM-COMGEA-11-0001, "Utilization of Expert Judgment in Regulatory Decision Making" (March 15, 2011).

[21] SRM-SECY-11-0024, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews" (May 11, 2011).

Resources	FY 2012	FY 2013 (Projected)	FY 2014 (Projected)
Resources Currently Requested for Future Level 3 PRA Activities			
FTE	2.0	2.0	N/A
Contract Support	$500k	$500k	N/A
Option 1: Maintain Status Quo – Continue Evolutionary Development of PRA Technology			
FTE	-2.0	-2.0	N/A
Contract Support	-$500k	-$500k	N/A
Option 2: Research to Address Identified Gaps Before Performing Future Level 3 PRAs			
FTE	0.0	0.0	N/A
Contract Support	$0k	$0k	N/A
Option 3: Full-Scope Comprehensive Site Level 3 PRA			
FTE	6.0	6.0	6.0
Contract Support	$1,500k	$1,500k	$1,500k

COORDINATION:

The Office of the Chief Financial Officer has reviewed this paper for resource implications and has no objections. The Office of the General Counsel has reviewed this paper and has no legal objection.

/RA by Michael F. Weber for/

R. W. Borchardt
Executive Director
for Operations

Enclosures:
1. Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities.
2. The Structure and Evolution of Probabilistic Risk Assessment and Risk-Informed Regulation.

Options for Proceeding with Future
Level 3 Probabilistic Risk Assessment Activities

PURPOSE

This document provides more detailed information on (1) the basis for originally proposing a scoping study to evaluate the feasibility of performing a new full-scope comprehensive site Level 3 probabilistic risk assessment (PRA) for a nuclear power plant (NPP)[1], (2) potential future uses for Level 3 PRAs, (3) three primary options for proceeding with future Level 3 PRA activities[2], and (4) the activities that supported development of items 2 and 3.

A separate document included as the second enclosure to the notation vote SECY paper provides more detailed information on the structure and evolution of PRA and risk-informed regulation that led to the staff's original proposal for a scoping study to evaluate the feasibility of performing a new full-scope comprehensive site Level 3 PRA.

[1] As used in this document and the SECY paper to which it is enclosed, a full-scope comprehensive site Level 3 PRA is a PRA that includes a quantitative assessment of the public risk from accidents involving all site reactor cores and spent nuclear fuel that can occur during any plant operating state, and that are caused by all initiating event hazards (internal events, fires, flooding, seismic events, and other site-specific external hazards).

[2] This document and the SECY paper to which it is enclosed distinguish between "Level 3 PRA activities" and "Level 3 PRAs." The latter refers to a PRA that includes specific technical elements or analyses to assess the public risk from a NPP, while the former refers to activities (e.g., research and development) specifically related to or in support of Level 3 PRAs.

Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities

BASIS FOR PROPOSING NEW LEVEL 3 PRA ACTIVITIES

In 1995, the Commission established the current framework for risk-informed regulation by issuing a PRA Policy Statement[3] that stated the use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art and in a manner that complements the U.S. Nuclear Regulatory Commission's (NRC's) deterministic approach and traditional defense-in-depth philosophy. In its approval, the Commission articulated its expectation that implementation of this policy would improve the regulatory process in three areas (1) through safety decisionmaking enhanced by the use of PRA insights, (2) through more efficient use of agency resources, and (3) through a reduction in unnecessary burdens on licensees.

Traditionally focused on accidents involving single-unit reactor cores, PRAs for NPPs can estimate risk metrics at three sequential levels or end states. A Level 1 PRA models system (plant and operator) response to various initiating events that challenge system operation to estimate reactor core damage frequency (CDF). A Level 2 PRA includes Level 1 PRA analyses and, in addition, models system and containment response to severe core damage accidents to estimate conditional containment failure probabilities, radioactive material release frequencies (e.g., large early release frequency [LERF]), and various source term characteristics. Finally, a Level 3 PRA includes Level 2 PRA analyses and, in addition, models the transport and dispersion of released radioactive materials to estimate various offsite radiological health and economic consequence measures. By combining radioactive material release frequencies from a Level 2 PRA with the offsite radiological consequences associated with each release, a Level 3 PRA estimates the public risk from all analyzed risk contributors[4]. Level 3 PRAs can provide valuable insights into the relative importance of various risk contributors to enhance regulatory decisionmaking and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety.

Using information from the last set of NRC-sponsored Level 3 PRAs conducted as part of the NUREG-1150 study[5], the staff determined that the reactor-specific risk metrics CDF and LERF can be used respectively as surrogates for the latent cancer risk and prompt fatality risk quantitative health objectives (QHOs) defined in the Commission's Safety Goal Policy Statement[6]. Therefore, instead of using Level 3 PRAs, the staff compares the results from Level 1 and limited-scope Level 2 PRAs to subsidiary numerical objectives based on CDF and LERF for regulatory decisionmaking involving plant-specific applications. Although Level 3 PRAs have since been performed to some extent within both the United States and international nuclear industries, the NRC has not sponsored development of a Level 3 PRA since NUREG-1150.

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[3] 60 FR 42622, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (August 16, 1995).

[4] As used in this SECY paper and its enclosures, risk contributors include: radiological sources (e.g., reactor core, spent nuclear fuel); initiating event hazards (e.g. internal events, fires, flooding, seismic events, other site-specific external hazards); plant operating states; accident sequences; failure of structures, systems, and components; and operator actions.

[5] NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" (December 1990).

[6] 51 FR 30028, "Safety Goals for the Operations of Nuclear Power Plants" (August 21, 1986).

The staff has identified several compelling reasons for proceeding with a new and more comprehensive site Level 3 PRA that can be organized into two broad categories (1) technical advances since NUREG-1150 and (2) additional scope considerations.

Technical Advances Since NUREG-1150

In the more than two decades since the last NRC-sponsored Level 3 PRAs were conducted as part of the NUREG-1150 study, numerous technical advances have been made that were not reflected in the NUREG-1150 PRA models. These technical advances can be organized into three categories (1) modifications to enhance NPP operational performance, safety, and security; (2) significantly improved understanding and modeling of severe accident phenomena; and (3) advances in PRA technology. Given the substantial role the NUREG-1150 results and risk insights have played in shaping the development and implementation of the current risk-informed regulatory framework, the potential impact of these advances warrants further investigation.

Modifications to Enhance NPP Operational Performance, Safety, and Security

PRA models should strive to be as realistic as practicable, representing the as-designed, as-built, and as-operated plant. Over the past two decades, the increased use of PRA results and insights by both the nuclear industry and the NRC has helped to improve NPP safety and operational flexibility and performance. These improvements have been realized in terms of observed reductions in the frequencies of the following types of events typically modeled in NPP PRAs: component unreliability (e.g., a pump failing to start or failing to run), component or train unavailability resulting from test or maintenance outages, special events covering operational issues (e.g., pump restarts injection valve reopening during unplanned demands), and initiating events.[7] In addition to the implementation of multiple risk-informed regulations, there have also been a number of modifications to plant design, operating and emergency procedures, and training, inspection, and maintenance practices. Finally, following the accident at Three Mile Island (TMI) and the terrorist attacks on September 11, 2011, the nuclear industry developed and implemented severe accident management and extensive damage mitigation strategies, respectively, to enhance both the safety and security of NPPs.

Notable examples of risk-informed regulations and guidance that have resulted in modifications to enhance NPP operational performance, safety, and security include:

- 10 CFR 50.44, Combustible gas control for nuclear power reactors
- 10 CFR 50.62, Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled NPPs
- 10 CFR 50.63, Loss of all alternating current power (Station blackout rule)
- 10 CFR 50.65, Requirements for monitoring the effectiveness of maintenance at NPPs (Maintenance rule)
- 10 CFR 50.54(hh)(2), Conditions of licenses – potential aircraft threat. These are also known as extensive damage mitigation guidelines (EDMGs) or B.5.b mitigation strategies[8].

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[7] NUREG/CR-6928, “Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants” (February 2007).

[8] EA-02-026, “Order for Interim Safeguards and Security Compensatory Measures” (February 25, 2002). Section B.5.b requires licensees to adopt mitigation strategies using readily available resources to maintain or restore core cooling, containment, and SFP cooling capabilities to cope with the loss of large areas of the facility due to large fires and explosions from any cause, including beyond-design-basis aircraft impacts. These requirements were formalized through rulemaking in 10 CFR 50.54(hh)(2).

- Severe Accident Management Guidelines (SAMGs)

Significantly Improved Understanding and Modeling of Severe Accident Phenomena
Insights gained from substantial research programs implemented after the TMI accident have significantly improved both our understanding of severe accident phenomena and the modeling of these phenomena using computer codes to support severe accident progression and containment response analyses as part of Level 2 and Level 3 PRAs.

Advances in PRA Technology
Similarly, insights gained from PRA-related research, advances in information and computer technology, and the acquisition of over 20 additional years of operating experience, have led to advances in PRA methods, models, tools, and data—collectively referred to as "PRA technology." Examples of important advances in PRA technology include improved modeling of severe accident phenomena; development of improved methods for common cause failure (CCF) analysis and human reliability analysis (HRA); improved analytical tools such as those used in the development and demonstration of state-of-the-art integrated modeling and analysis of severe accident progression and offsite radiological consequences in the State-of-the-Art Reactor Consequence Analysis (SOARCA) Project; and improved quality and quantity of data for initiating events, component failures, and operator errors. These advances in our knowledge and PRA technology through research and acquired operating experience should result in improved methods, models, tools, and data when compared to the NUREG-1150s; this in turn should lead to an associated reduction in the epistemic or "state-of-knowledge" uncertainties that can significantly impact the interpretation and use of PRA results and risk insights.

<u>Updated Seismic Hazard Data</u>
Generic Issue (GI)-199, "Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants," recently investigated the safety and risk implications of updated earthquake-related data and models. These data and models suggested that the probability of earthquake ground shaking above the seismic design basis for some NPPs in the Central and Eastern United States (CEUS) is still low, but larger than previous estimates.

Key messages from the GI-199 Safety/Risk Assessment (ML100270582) included:

(1) **Operating NPPs are safe.** Plants have adequate safety margin for seismic issues. The NRC's Safety/Risk Assessment confirms that overall seismic risk estimates remain small and that adequate protection is maintained.

(2) **Though still small, some seismic hazard estimates have increased.** Updates to seismic data and models indicate increased seismic hazard estimates for some operating NPP sites in the CEUS.

(3) **Assessment of GI-199 will continue.** Plants are safe, but the NRC has separate criteria for evaluating whether plant improvements may be imposed. The NRC's Safety/Risk Assessment used readily available information and found that for about one-quarter of the currently operating plants, the estimated CDF change is large enough to warrant further attention. Action may include obtaining additional, updated information and developing methods to determine if plant improvements to reduce seismic risk are warranted.

These insights gained from the GI-199 Safety/Risk Assessment suggest a need for further evaluating the relative contribution of the seismic hazard to the public risk from all analyzed risk contributors associated with NPPs.

Additional Scope Considerations

In addition to these technical advances since NUREG-1150, the staff has identified additional scope considerations not previously considered that could be addressed by performing a new and more comprehensive site Level 3 PRA. Examples of these additional scope considerations include (1) consideration of multi-unit[9] site effects and (2) consideration of other site radiological sources (e.g., spent fuel pools [SFPs], dry storage casks, and multiple units). Each of these areas is explored in more detail below. Before doing so, however, some of the important limitations of previous Level 3 PRAs and current PRAs used to support regulatory applications that could be addressed by expanding the scope to include additional considerations are reviewed.

Scope Limitations of the NUREG-1150 PRAs

The NUREG-1150 PRAs were limited in scope to the assessment of single-unit reactor accidents initiated primarily by internal events occurring during at-power operations, with only a partial treatment of fires and seismic events for two of the five analyzed plants (Surry and Peach Bottom). Although a later study evaluated the risk associated with accident sequences occurring during low-power/shutdown operations for two of the five analyzed plants (Grand Gulf[10] and Surry[11]), this study examined only one plant operating state (POS) in detail and was unable to provide an integrated perspective with insights into the relative importance of multiple POSs to risk. In addition, the NUREG-1150 PRAs did not include an assessment of accidents involving other site radiological sources such as spent fuel pools, dry storage casks, and other units on site (including additional reactor cores and spent fuel pools). These considerations are discussed in more detail below.

Current Use of Limited-Scope PRAs for Regulatory Applications

Although Regulatory Guide (RG) 1.174[12] states that the CDF and LERF acceptance guidelines are intended for comparison with the results of a full-scope PRA that includes all risk contributors, it does allow for the use of limited-scope PRAs. When a limited-scope PRA is used, the contribution of out-of-scope items to risk must be assessed based on the margin between the PRA results and the acceptance guidelines.

However, with qualitative analyses of varying degrees of rigor being submitted to support scope limitations, guidance is needed for the staff to assess the impact of these scope limitations on conclusions that are made. Among others, a study sponsored by the Advisory Committee on

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[9] As used in this document and the SECY paper to which it is enclosed, a unit refers to a reactor core and, if applicable, an associated spent fuel pool.

[10] NUREG/CR-6143, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1" (March 1995).

[11] NUREG/CR-6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1" (July 1994).

[12] Regulatory Guide 1.174, Rev. 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (May 2011).

Reactor Safeguards (ACRS) to assess the agency's need for improved PRA technology to risk inform its regulations[13] identified the following point as needing further investigation:

> "While there are valid technical arguments that can be made to justify the exclusion of some portions of a full-scope PRA model for risk-informed regulation, there are resources that must be continually applied by the licensee and the NRC to check the validity of the risk-informed decisions in light of the use of an incomplete PRA model. At some point, it is reasonable to ask whether these additional resources are small or large in relation to the use of a full-scope PRA to start with."

Consideration of Multi-Unit Site Effects

Because the Commission's safety goals, QHOs, and subsidiary numerical objectives are applied on a per reactor basis, most PRAs developed to date do not explicitly consider multi-unit accidents in which initiating events lead to reactor core damage in multiple units at the same site. Current PRA models therefore do not generally identify and address dependencies between systems at multi-unit sites, particularly those with highly interdependent support systems involving systems and subsystems that are shared by multiple units.

To understand the contribution of these multi-unit effects to the risk associated with a NPP, PRA models need to be enhanced to include both initiating events that might simultaneously impact multiple units and equipment and human action dependencies in responding to multi-unit accidents.

Consideration of Other Site Radiological Sources

To be complete, estimation of total site accident risk should also include an assessment of the risk from accidents involving other site radiological sources, to include spent nuclear fuel.

In summary, the NRC has never sponsored a site Level 3 PRA that includes an assessment of not only accidents involving the reactor core of a single unit, but also accidents involving other site radiological sources such as spent fuel pools, dry storage casks, and other units on site (including additional reactor cores and spent fuel pools). The incorporation of these technical advances and additional scope considerations into a new full-scope comprehensive site Level 3 PRA could yield new and improved risk insights to enhance regulatory decisionmaking and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety.

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[13] NUREG/CR-6813, "Issues and Recommendations for Advancement of PRA Technology In Risk-Informed Decision Making" (April 2003).

COMMISSION TASKING

During the Annual Commission Briefing on Research Programs, Performance, and Future Plans on February 18, 2010, the staff proposed a scoping study to evaluate the feasibility of performing a new full-scope comprehensive site Level 3 PRA.

In SRM M100218 (ML100780578) dated March 19, 2010, the Commission expressed conditional support for Level 3 PRA related activities and directed the staff to (1) continue internal coordination efforts and engage external stakeholders in formulating a plan and scope for future actions and (2) provide the Commission with various options for proceeding that include costs and perspectives on future uses for Level 3 PRAs.

This document and the notation vote SECY paper to which it is enclosed were developed in response to the SRM. The remainder of this document discusses potential future uses for Level 3 PRAs and three primary options for proceeding with future Level 3 PRA activities that were developed by the staff through multiple interactions with stakeholders.

APPROACH TO NEW LEVEL 3 PRA ACTIVITIES
In response to SRM M100218 and to optimize cost-benefit considerations by focusing NRC resources, the staff developed a three-phased approach to planning and conducting future Level 3 PRA activities.

The first phase consisted of a scoping study that began in April 2010 and ended upon submission of this SECY paper to the Commission. This scoping study was conducted by staff from the Office of Nuclear Regulatory Research (RES) with support from representatives from the following NRC program offices: Office of Nuclear Material Safety and Safeguards (NMSS), Office of New Reactors (NRO), Office of Nuclear Reactor Regulation (NRR), and Office of Nuclear Security and Incident Response (NSIR). The objectives and activities associated with this scoping study are discussed in more detail below.

The second phase would consist of proceeding with either one of the options developed by the staff as part of the scoping study or any other option directed by the Commission following submission of the notation vote SECY paper.

Based on the results and insights from the second phase, the staff would then assess the need for follow-on Level 3 PRA activities and then provide the Commission with additional options and recommendations for proceeding.

LEVEL 3 PRA SCOPING STUDY

Objectives

Based on Commission tasking in SRM M100218, the staff identified the following main objectives for the Level 3 PRA scoping study:

(1) To identify potential future uses for Level 3 PRAs;

(2) To develop various options for proceeding with future Level 3 PRA activities that include objectives, scope, PRA technology to be used, site selection considerations[14], and resource estimates;

(3) To determine the feasibility of proceeding with each of the developed options;

(4) To continue internal coordination efforts to identify the staff's recommendation for proceeding; and

(5) To engage external stakeholders to obtain their views on the staff's approach, potential future uses for Level 3 PRAs, options for proceeding with future Level 3 PRA activities, and recommendation for proceeding.

Internal Coordination Activities

Throughout the scoping study, the staff participated in numerous internal coordination activities to develop its approach, identify potential future uses for Level 3 PRAs, develop options for proceeding with future Level 3 PRA activities, and identify its recommendation for proceeding. These activities included workshops, coordination and alignment meetings, and internal stakeholder briefings.

Brainstorming Workshop

The scoping study began with a brainstorming workshop on April 28, 2010 that was attended by RES staff and managers, as well as select staff from NMSS, NRO, NRR, and NSIR. The primary objectives of this workshop were to (1) provide participants with the background and vision for new Level 3 PRA activities, (2) identify scoping issues associated with various technical elements of new Level 3 PRAs, (3) identify potential uses for future Level 3 PRAs, and (4) identify technical working groups for the scoping study and next steps for moving forward.

As a result of the workshop, the following six technical working groups comprised of staff from RES, NMSS, NRO, NRR, and NSIR were established to accomplish the scoping study objectives for specific Level 3 PRA technical elements that were viewed as particularly complex and challenging (1) Level 1 PRA and Interface to Level 2 PRA, (2) Level 2 PRA and Interface to Level 3 PRA, (3) Other (than internal events) Hazard Groups PRA, (4) Spent Fuel and Non-Reactor PRA, (5) Human Reliability Analysis, and (6) 21st Century PRA Documentation.

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[14] Because the Commission expressed only conditional support and directed the staff to provide various options for proceeding, the staff did not include selection of a site to participate in a future Level 3 PRA as one of the objectives of the scoping study. Instead, the staff identified various site selection considerations related to the quality and availability of relevant information that could impact the cost of a future Level 3 PRA. These considerations, which are provided later in this enclosure, can inform future site selection activities if the Commission directs the staff to proceed with a Level 3 PRA.

An Integration and Oversight working group was also established to oversee technical working group activities and to integrate the options developed by each technical working group to develop high-level options and a specific recommendation for proceeding. This group created a working group charter to provide the technical working groups with objectives, deliverables, working group roles and responsibilities, and guidance for developing various scoping options.

Alignment Workshop

On July 27, 2010, working group leaders and select staff and managers from RES, NMSS, NRO, and NRR participated in an alignment workshop to (1) obtain an overview of the scoping options being considered and developed by the technical working groups; (2) identify ongoing or planned research that supports and has cost implications for these scoping options; (3) identify and discuss working group interface issues; (4) ensure initial alignment on key messages among working group leaders, senior managers, and other representatives from offices who will be involved in the Commission paper concurrence process; and (5) consider site selection issues and options for engaging external stakeholders.

Coordination and Alignment Meetings

Throughout the scoping study, the staff held numerous coordination meetings with senior managers and technical staff in RES, NMSS, NRO, NRR, and NSIR to further develop and refine the various options for proceeding with future Level 3 PRA activities, including costs and potential future uses for Level 3 PRAs, and to identify the staff's recommendation for proceeding.

In accordance with guidance provided by the Office of the Executive Director for Operations (OEDO) on the process for developing SECY papers[15], the staff also participated in multiple alignment meetings to ensure senior management and technical staff were in agreement on expectations, scope, and key messages to be communicated in the notation vote SECY paper to which this document is attached.

Internal Stakeholder Briefings

Throughout the scoping study, the staff provided multiple briefings to various internal stakeholders. The purposes of these briefings varied depending on the audience, but in general included stimulating interest in the initiative, sharing information about the staff's current thinking, answering stakeholder questions, and seeking stakeholder feedback. Example briefings include NMSS/NSIR staff briefing on April 12, 2010; Senior Reactor Analyst (SRA) monthly call on August 16, 2011; SRA Counterpart Meeting on May 26, 2011; and briefing of the Chairman's task force for developing options for a more holistic risk-informed, performance-based regulatory approach[16] on May 31, 2011.

External Stakeholder Engagement Activities

In addition to internal coordination activities, the staff participated in numerous external stakeholder engagement activities during the scoping study. These activities included: ACRS interactions, Regulatory Information Conference (RIC) presentations, and a Category 2 public meeting with representatives from nuclear industry, vendor, research, interest group, and public media organizations.

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[15] OEDO Notice 2010-0380-01, "SECY Paper Development Process" (March 17, 2010).

[16] Memorandum from G.B. Jaczko to R.W. Borchardt, "Assessment of Options for More Holistic Risk-Informed, Performance-Based Regulatory Approach" (February 11, 2011).

ACRS Interactions

The staff interacted with the ACRS on three separate occasions. The first briefing occurred during a November 17, 2010 meeting of the ACRS Subcommittee on Reliability and PRA in which the staff presented its approach to planning for future Level 3 PRA activities. The second briefing occurred during a May 11, 2011 meeting of the ACRS Subcommittee on Reliability and PRA in which the staff presented its identified potential uses for future Level 3 PRAs, developed options for proceeding with future Level 3 PRA activities, and its recommendation for proceeding. The third and final briefing occurred during a June 8, 2011 meeting of the ACRS Full Committee in which the staff once again presented its identified potential uses for future Level 3 PRAs, developed options for proceeding with future Level 3 PRA activities, and its recommendation for proceeding. The goal of this final briefing was to obtain ACRS support for the staff's recommendation.

Following the briefing of the Full Committee, the ACRS recommended the staff (1) develop a phased approach and schedule that would enable the staff to complete a new full-scope comprehensive site Level 3 PRA while minimizing the near-term resource impact; (2) take maximum advantage of existing PRA technology; and (3) actively engage the participation of industry. These recommendations are consistent with the staff's plans for proceeding, if the Commission directs the staff to proceed with such a study.[17]

RIC Presentations

The NRC took advantage of unique opportunities to engage with both internal and external stakeholders at the 2010 and 2011 RIC. At the 2010 RIC, the staff introduced stakeholders to this new Level 3 PRA initiative by presenting the envisioned approach, objectives, and scope in the "Current Topics in Probabilistic Risk Analysis" technical session. At the 2011 RIC, the staff provided stakeholders with updated information and encouraged their engagement and participation in the subsequent ACRS interactions and the Category 2 public meeting discussed below by developing a poster presentation and by presenting in the "Current Topics in Probabilistic Risk Analysis" technical session. During and after the 2011 RIC, the staff received a number of stakeholder questions related to this new Level 3 PRA initiative. The staff developed responses to each of these questions and posted them to the external RIC website for stakeholders to review.

Category 2 Public Meeting

A Category 2 public meeting was held on April 11, 2011 at NRC Headquarters to obtain stakeholder views on options for proceeding with future Level 3 PRA activities. Key external stakeholders who were specifically invited to attend and who participated in the public meeting included representatives from the following organizations: the Nuclear Energy Institute (NEI), the Electric Power Research Institute, Inc. (EPRI), and the Union of Concerned Scientists (UCS). Representatives from other nuclear industry, vendor, research, interest group, and public media organizations also participated.

In general, meeting participants supported the options developed by the staff for proceeding with future Level 3 PRA activities. In particular, meeting participants supported a recommendation to proceed with a new full-scope comprehensive site Level 3 PRA, but expressed some concern about the potential scope, cost, and schedule of such a study. Some participants offered comments related to specific aspects of technical elements of a Level 3 PRA that the NRC should consider if a new full-scope comprehensive site Level 3 PRA is

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[17] Letter from Said Abdel-Khalik to The Honorable Gregory B. Jaczko, "Draft SECY Paper, "Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities"" (June 22, 2011).

planned. A public meeting summary, including a list of meeting participants and a copy of the meeting presentation slides handout is publicly available in the Agencywide Documents Access and Management System (ADAMS) at ML111400179.

POTENTIAL FUTURE USES FOR LEVEL 3 PRAs

In identifying potential future uses for Level 3 PRAs, a logical first step was to identify how the results and risk insights from the NUREG-1150 PRAs were used. In addition, the staff considered potential enhancements that could be made to the use of PRA in the existing risk-informed regulatory framework. In this way, the staff developed the following set of seven potential uses. These potential uses are meant to apply to future Level 3 PRAs in general, and not specifically to the full-scope comprehensive site Level 3 PRA proposed as Option 3; the use of this specific PRA would ultimately depend on its scope and applicability to the larger population of NPP sites.

Confirm the Acceptability of the Agency's Current Use of PRA in Risk-informed Regulatory Decisionmaking

Future Level 3 PRAs could be used to assess the agency's current use of PRA in risk-informed regulatory decisionmaking. Examples include the use of Level 1 and limited-scope Level 2 PRAs to support regulatory applications, and the use of RG 1.174 subsidiary numerical objectives based on the reactor-specific risk metrics CDF and LERF that were originally developed and validated using NUREG-1150 information.

Verify or Revise Regulatory Requirements and Guidance

Future Level 3 PRAs could be used to either verify or revise regulatory requirements and guidance, particularly those based on the last NRC-sponsored Level 3 PRAs that were conducted as part of the NUREG-1150 study. In addition to the previously discussed RG 1.174 acceptance guidelines based on CDF and LERF that are used by the staff in regulatory decisionmaking involving plant-specific applications, this would include the regulatory analysis guidelines[18] and technical evaluation handbook[19] used by the staff to evaluate proposed backfits[20] to determine, among other things, whether the benefits associated with a proposed regulatory action are commensurate with the cost. The NRC performs regulatory analyses to support numerous NRC actions that affect its reactor licensees. The regulatory analysis guidelines and handbook contain a number of policy decisions that have broad implications for the NRC and its licensees, including the use of safety goal evaluations, a $2000 per person-rem conversion factor, and criteria for the treatment of individual requirements.

Support Specific Risk-Informed Regulatory Applications

Future Level 3 PRAs could be used to provide support for a variety of specific risk-informed regulatory applications. Examples include providing the technical basis for risk-informing the regulation of spent fuel storage and handling, siting, and emergency preparedness; and focusing the Reactor Oversight Process (ROP), including the NRC's inspection program.

Develop and Pilot Test PRA Technology, Standards, and Guidance

Future Level 3 PRAs could be used to develop and pilot test new PRA technology (e.g., methods, models, and tools) developed to obtain new and improved risk insights; consensus PRA standards; and regulatory guidance to ensure requirements are clear, understandable, and achieve consistency.

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[18] NUREG/BR-0058, Rev. 4, "Regulatory Analysis Guidelines of the U.S. Nuclear Regulatory Commission" (September 2004).

[19] NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook" (January 1997).

[20] Title 10 Section 50.109, of the *Code of Federal Regulations* (10 CFR 50.109). Backfitting.

Prioritize Generic Issues and Nuclear Safety Research Programs
Future Level 3 PRAs could also be used to inform the prioritization and resolution of GIs and the prioritization of nuclear safety research programs by focusing limited agency resources on issues most directly related to the agency's mission to protect public health and safety. This was one of the identified uses of the NUREG-1150 PRA results and risk insights.

Develop In-House PRA Technical Capability and Support PRA Knowledge Management and Risk Communication Activities
Future Level 3 PRAs could be used to support a variety of PRA staffing, knowledge management, and risk communication activities. Development of future Level 3 PRAs within the NRC could help develop first-hand knowledge about PRA and the technical skills needed for performing and reviewing PRAs. To support knowledge management activities, they could provide the technical basis for updating training materials for PRA developers, reviewers, and users. In addition, by using modern information technology to document the relevant assumptions, decisions, methods, models, tools, and data, future Level 3 PRAs can provide readily accessible information to support potential future needs. In addition to improving internal risk communication by improving PRA training and making PRA information more accessible, future Level 3 PRAs with improved documentation can be used to enhance external risk communication by facilitating external stakeholder understanding of not only the relative importance of various risk contributors to public risk, but also the underlying assumptions and limitations affecting the results and risk insights.

Support Future Risk-Informed Licensing of New and Advanced Reactor Designs
Future Level 3 PRAs could be used to support the future risk-informed licensing of new and advanced reactor designs. First, in its Policy Statement on the Regulation of Advanced Reactors[21], the Commission stated its intention to "improve the licensing environment for advanced nuclear power reactors to minimize complexity and uncertainty in the regulatory process." The staff noted in its Advanced Reactor Research Plan (ML082530184) that a risk-informed regulatory structure applied to license and regulate advanced reactors, regardless of their technology, could enhance the effectiveness, efficiency, and predictability of future plant licensing. In NUREG-1860[22], the staff documented the results of a study that was conducted to establish the feasibility of developing a risk-informed and performance-based regulatory framework for the licensing of future NPPs that could be used to develop a set of regulatory requirements that would serve as an alternative to 10 CFR 50. This framework was envisioned to have the following potential advantages:

(1) It would require a broader use of design-specific risk information in establishing the licensing basis, thus better focusing on those items most important to safety for that design;

(2) It would stress the use of performance as the metric for acceptability; and

(3) It could be written to be applicable to any reactor technology ("technology neutral'), thus avoiding the time consuming and less predictable process of reviewing non-light water

[21] 73 FR 60612, "Policy Statement on the Regulation of Advanced Reactors" (October 14, 2008).

[22] NUREG-1860, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing" (December 2007).

reactor (LWR) designs against the LWR-oriented regulations in 10 CFR 50 and making decisions on a case-by-case basis.

In this concept of a "technology neutral framework," a design-specific, full-scope Level 3 PRA would be used to identify licensing basis events (LBEs) by comparing the frequencies and consequences of all possible event scenarios with a frequency-consequence (F-C) curve established by various site boundary radiation dose limits. The LBEs, whose purpose is principally similar to that of the design basis accidents in the current regulatory framework, are selected from those PRA event sequences whose frequencies and consequences approach the F-C curve. This process is further used to inform defense-in-depth (including safety margin) requirements and the safety categorization of structures, systems, and components (SSCs).

In the SRM to SECY-07-0101[23], the Commission stated that the staff should publish the technology-neutral framework and its concepts should be tested on an actual design. Although the Commission indicated in this SRM that the Pebble Bed Modular Reactor (PBMR) design review would be a logical choice, the testing of the framework has not yet occurred. Future Level 3 PRAs could be used for this purpose.

In addition to pilot testing the "technology neutral framework," future Level 3 PRAs could be used to inform the staff's follow-on activities related to (1) resolving issues with small modular reactor (SMR) designs[24], (2) using risk insights to enhance the safety focus of SMR reviews[25], and (3) modifying risk-informed regulatory guidance for new reactors[26]. Although future Level 3 PRAs would not be developed in time to inform the staff's current activities related to these efforts, they could be used to inform related follow-on activities.

[23] SECY-07-0101, "Staff Recommendations Regarding a Risk-Informed and Performance-Based Revision to 10 CFR Part 50 (RIN 3150-AH81)" (June 14, 2007).

[24] SECY-10-0034, "Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs" (March 28, 2010).

[25] SECY-11-0024, "Use of Risk Insights to Enhance the Safety Focus of Small Modular Reactor Reviews" (February 18, 2011).

[26] SECY-10-0121, "Modifying the Risk-Informed Regulatory Guidance for New Reactors" (September 14, 2010).

OPTIONS FOR PROCEEDING WITH FUTURE LEVEL 3 PRA ACTIVITIES

This section presents detailed descriptions of the three primary options deemed by the staff to best frame the choices from a feasibility and cost-benefit perspective and their relative advantages and disadvantages.

Examples of other options that were considered by the staff through participation in the previously discussed scoping study activities are provided in the notation vote SECY paper to which this document is enclosed.

Option 1: Maintain Status Quo – Continue Evolutionary Development of PRA Technology
This option maintains the status quo in ongoing activities related to the development and implementation of PRA technology and risk-informed regulation. Ongoing and planned research to develop and improve upon existing PRA methods, models, tools, and data would continue on a resource-available basis as driven by program office user need requests (UNRs) and the agency's long term research plan (LTRP). The staff also would continue to monitor relevant developments within both the United States and international nuclear industries.

As part of its strategic LTRP efforts, the staff has identified Level 2 and Level 3 PRA as areas that would benefit from examination of advanced methods, and is performing limited research in these areas. For example, under this program, a scoping study to evaluate both methodological and implementation-oriented issues associated with the advancement of Level 2 and Level 3 PRA modeling techniques was recently completed. This study resulted in the development of a spectrum of modeling approaches, which included: modified traditional approaches, hybrid event tree approaches, dynamic event tree approaches, and sampling-based simulation approaches[27]. As a result, a new phase of work that focuses on the dynamic event tree approach using the MELCOR Severe Accident Analysis Code in conjunction with a dynamic operator response model has begun. The initial methods development, including application of the approach to a demonstration problem, is scheduled to be completed by the end of calendar year 2011.

Advantages

- Is consistent with the current fiscal climate by focusing limited staff and contract support resources on mission-critical work driven by program office UNRs, Commission tasking, and the agency's LTRP.

Disadvantages

- Insights that could be gained by conducting a full-scope comprehensive site Level 3 PRA to enhance regulatory decisionmaking would not be realized.

- Can result in inconsistent and more costly treatment of potential future issues by developing the necessary PRA technology on an ad-hoc basis.

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[27] Helton, D. "Scoping Study on Advancing Modeling Techniques for Level 2/3 PRA" (May 2009). Available in ADAMS at ML091320454.

Option 2: Conduct Focused Research to Address Identified Gaps in Existing PRA Technology Before Performing a Full-Scope Comprehensive Site Level 3 PRA

This option involves near-term focused research aimed at addressing identified gaps in existing PRA technology over the next 2 years. These technical gaps are related to the expanded scope and the differing degrees of sophistication in the existing PRA technology used to analyze the risk from various risk contributors.

This option was developed with the understanding that reallocating resources to develop a new full-scope comprehensive site Level 3 PRA would be particularly challenging because of mission-critical work already assigned to a limited number of qualified risk analysts. Moreover, a decision on whether to proceed with such a study would benefit from better understanding the recommendations and Commission tasking from multiple task forces (e.g., the Chairman's task force to develop options for a more holistic risk-informed, performance-based regulatory approach and the near-term task force to conduct methodical and systematic reviews of our current processes and regulations in response to the recent events in Japan[28]).

Objective

The primary objective of this research would be to ensure that important technical gaps related to the expanded scope and the differing degrees of sophistication in the existing PRA technology used to analyze the risk from various risk contributors are addressed before developing a new full-scope comprehensive site Level 3 PRA.

[28] Tasking Memorandum – COMGBJ-11-0002, "NRC Actions Following the Events in Japan" (March 23, 2011).

Advantages

- Focuses limited available staff and contract support resources on mission-critical work driven by program office UNRs, Commission tasking, and the agency's LTRP.

- Focuses additional staff and contract support resources that have already been requested to support future Level 3 PRA activities on research needed to address identified gaps in existing PRA technology.

- Produces results and insights that would advance the state-of-practice in specific PRA technical elements and thereby enhance NRC's PRA capability in those technical areas.

Disadvantages

- Delays insights that could be gained by conducting a full-scope comprehensive site Level 3 PRA to enhance regulatory decisionmaking.

Scope

Examples of gaps in existing PRA technology that need to be addressed either prior to or in parallel with conducting future Level 3 PRAs include:

Modeling of Consequential (Linked) Multiple Initiating Events

Current PRA models do not include scenarios in which multiple, linked initiating events occur either simultaneously or close in time with respect to overall mission time such that a second initiating event occurs while the plant is still responding to the first. Methods for incorporating these types of scenarios into current PRA models need to be investigated.

Modeling of Multi-Unit Dependencies

Because the Commission's safety goals, QHOs, and subsidiary numerical objectives are applied on a per reactor basis, most PRAs developed to date do not explicitly consider multi-unit accidents in which initiating events lead to reactor core damage in multiple units at the same site. Current PRA models therefore do not appropriately identify and address dependencies between systems at multi-unit sites, particularly those with highly convoluted support system dependencies involving systems and subsystems that are shared by multiple units.

To understand the contribution of these multi-unit effects to the risk associated with a NPP, PRA models need to be enhanced to include both initiating events that might simultaneously impact multiple units and equipment and human action dependencies in responding to multi-unit accidents.

Post-Core Damage and External Events HRA Modeling

Severe Accident Management Guidelines (SAMGs)

In response to the TMI accident, the nuclear industry developed and implemented SAMGs to provide tools and strategies for managing the in-plant aspects and mitigating the results of a severe accident. The overall goal of SAMGs is to terminate emergency conditions by (1) returning the reactor core to controlled and stable state, (2) maintaining or returning the containment to a controlled and stable state, and (3) terminating any fission product releases.

The following three groups of individuals use SAMGs in the event of a severe accident leading to core damage:

(1) **Evaluators.** Evaluators are members of the plant evaluation team that are responsible for diagnosing plant conditions, evaluating the impacts of potential strategies, and assessing the effectiveness of implemented strategies.

(2) **Implementors.** Implementors are typically plant operators that are responsible for monitoring plant indications, operating equipment, and communicating with evaluators and decision makers.

(3) **Decision Makers.** Decision makers are typically plant managers or technical directors who are responsible for analyzing information and recommendations provided by both implementors and evaluators and for deciding which strategies to implement.

From a post-core damage HRA modeling perspective, the use of SAMGs presents a unique challenge. In Rasmussen's Cognitive Taxonomy, the emergency operating procedures (EOPs) used by operators to prevent core damage are "rule-based," and therefore allow for identification of the best course of action for any set of conditions by simply following procedures, However, SAMGs are "knowledge-based," and therefore require evaluators to use their knowledge and problem solving skills to identify the least-bad course of action in unfamiliar severe accident conditions. In addition, almost all of the SAMG strategies to mitigate the effects of one problem result in adverse effects on another problem. Evaluators must therefore make risk-benefit decisions when considering different strategies. Since the most appropriate response to a given condition cannot be determined in advance, the definition of what constitutes a failure and the identification of post-core damage human failure events or recovery actions that can be credited in the PRA model presents a unique challenge that needs to be addressed if a site Level 3 PRA model is to be developed.

Extensive Damage Mitigation Guidelines (EDMGs)
Following the terrorist attacks on September 11, 2011, the NRC required licensees to implement EDMGs described in Title 10, Section 50.54(hh). Much like SAMGs, the definition of what constitutes a failure and the identification of human failure events or recovery actions that can be credited in the PRA model presents a unique challenge that needs to be addressed if a site Level 3 PRA model is to be developed.

External Events HRA
In addition to addressing challenges associated with the modeling of SAMGs and EDMGs, research into the modeling of human actions in response to various external events (e.g., seismic events, external flooding) is needed.

<u>Spent Fuel PRA Technology</u>
Process areas not related to reactor core operations that can contribute to nuclear site accident risk include those associated with onsite nuclear spent fuel handling and storage. Although limited PRA models for quantitatively evaluating the risk of accidents involving spent fuel pools and dry cask storage exist, additional or significantly improved PRA technology must be developed to enable a meaningful comparison and relative ranking of these process area risk contributors as part of a comprehensive site Level 3 PRA. Example spent fuel PRA areas for improvement include: success criteria determination, HRA, accident phenomena, and source term analysis.

<u>Modeling of Aqueous Transport and Dispersion of Radioactive Materials</u>
As demonstrated by the recent events in Japan, certain accident scenarios can result in large volumes of contaminated water being generated by emergency measures to cool the reactor cores and SFPs, with yet to be determined offsite radiological consequences. To determine the relative risk significance of these types of scenarios, a Level 3 PRA must be capable of modeling and analyzing the aqueous transport and dispersion of radioactive materials through surface water, sediments, soils, and groundwater. Existing PRA analytical tools do not have this capability. Research is therefore needed to identify or develop methods, models, and tools that can be used to simulate geochemical speciation and transport of dissolved radionuclides in surface water, sediments, soils, and groundwater.

<u>Level 2 and Level 3 PRA Uncertainty Analysis</u>
Although guidance on the process for identifying and characterizing key sources of uncertainty exists[29], research is needed to identify the key sources of uncertainty in Level 2 and Level 3 PRA analyses and to develop specific methods for propagating uncertainty through the Level 2 and Level 3 PRA analyses.

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[29] NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (March 2009).

<u>Option 3: Conduct a Full-Scope Comprehensive Site Level 3 PRA</u>

This option involves planning for and performing a new full-scope comprehensive site Level 3 PRA for an operating NPP. Research identified in Option 2 also would be conducted as part of this option, but on an accelerated schedule to support the completion of a full-scope comprehensive site Level 3 PRA within 3 years. Option 2 and Option 3 differ only in terms of timing, sequencing, near-term use of resources, and relative advantages and disadvantages. In addition, selection of Option 2 would require separate Commission direction in the future before proceeding with a new full-scope comprehensive site Level 3 PRA.

Objectives

The staff has identified the following four high-level objectives for a new full-scope comprehensive site Level 3 PRA for an operating NPP:

(1) Extract new and improved risk insights to enhance regulatory decisionmaking and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety by:

   a. expanding the PRA scope to include an assessment of the risk from accidents involving spent fuel and multiple units,

   b. incorporating technical advances since NUREG-1150, and

   c. using a more integrated and consistent analysis approach to enable a meaningful comparison and relative ranking of all analyzed site risk contributors

(2) Enhance PRA capability, expertise, and documentation by improving upon existing analytical tools, by:

   a. providing training opportunities for staff and contractors, and

   b. using improved documentation practices and current information technology to make PRA information more accessible, retrievable, and understandable.

(3) Demonstrate the technical feasibility and evaluate the realistic cost of developing new Level 3 PRAs by leveraging both existing analytical tools and ongoing or planned relevant research, where appropriate, rather than developing entirely new models.

Advantages

- Provides new and improved risk insights earlier to enhance regulatory decisionmaking and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety.

- Enhances PRA capability, expertise, and documentation earlier to address potential future issues.

Disadvantages

- Is resource-intensive, requiring more staff and contract support resources than currently budgeted.

- Requires reallocation of qualified risk analysts from other ongoing important activities, potentially resulting in delays to reviews of National Fire Protection Association (NFPA) Standard 805[30] license amendments, refinement of Standardized Plant Analysis Risk (SPAR) models, and reviews of PRAs in support of combined operating license applications. A more detailed resource discussion, including the potential implications of selecting Option 3, is provided in the notation vote SECY paper to which this document is enclosed.

[30] NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants."

PRA Scope

The scope of this PRA would include (1) site radiological sources—all reactor cores, spent fuel pools, and dry storage casks on site; (2) initiating event hazards—internal events, fires, flooding, seismic events, and other site-specific external hazards; (3) POSs—at-power and low-power/shutdown. The only factors specifically excluded from the scope would be radiological sources involving fresh nuclear fuel and radiological waste, and initiating events involving deliberate malevolent acts (e.g., terrorism and sabotage). The table below illustrates the options included for each factor considered in the analysis.

Factor	Options Included in Full-Scope Comprehensive Site Level 3 PRA
Radiological hazards	Reactor core(s) Spent nuclear fuel (spent fuel pools and dry storage casks)
Population exposed to hazards	Offsite population
Initiating event hazard groups	Internal hazards • Internal events (transients, loss-of-coolant accidents) • Internal floods • Internal fires
	External hazards • Seismic events (earthquakes) • Other site-specific external hazards (e.g., high winds, external flooding)
Plant operating states	At-Power Low-Power/Shutdown
End state/Risk metrics	Level 1 PRA: Core damage frequency* Level 2 PRA: Large early release frequency* Level 3 PRA: Number of early fatalities Number of early injuries Number of latent cancer fatalities Population dose (person-rem) at various locations Individual early fatality risk defined in QHO Individual latent cancer fatality risk defined in QHO Economic costs of mitigation actions**

* Although the Level 3 PRA will be used to estimate the public risk in terms of a variety of consequence measures, it is envisioned that the CDF and LERF risk metrics will be computed in intermediate steps to obtain near-term benefit in support of existing risk-informed regulatory applications.

** Although the staff previously considered the possibility of developing additional safety goals based on the risk of land contamination and overall societal impact, based on significant weaknesses in the analytical tools at the time, the staff recommended not pursuing this effort[31]. If the staff were to perform a new full-scope comprehensive site Level 3 PRA, it would plan on estimating economic risk associated with mitigation actions such as land interdiction, condemnation, and decontamination. These calculations could easily be performed by existing analytical tools described in more detail below at relatively little, if any, additional cost. By doing so, the staff would not be proposing to use this information for regulatory decisionmaking; instead, it would be used as an additional source of risk insights.

[31] SECY-00-0077, "Modifications to the Reactor Safety Goal Policy Statement" (March 30, 2000).

PRA Technology
Consistent with the above objectives to enhance PRA capability and to demonstrate the technical feasibility and evaluate the cost of developing new Level 3 PRAs by leveraging both existing analytical tools and ongoing or planned relevant research, the staff envisions using the following existing PRA technology as part of a new full-scope comprehensive site Level 3 PRA:

SPAR Models
The staff uses SPAR models in support of risk-informed activities related to the inspection program, incident investigation program, license amendment reviews, performance indicator verification, accident sequence precursor (ASP) program, GIs, and special studies. These tools also support and provide rigorous and peer-reviewed evaluations of operating experience, thereby demonstrating the agency's ability to analyze operating experience independently of licensees' risk assessments and enhancing the technical credibility of the agency.

The SPAR models integrate systems analysis, accident scenarios, component failure likelihoods, and HRA into a coherent model that reflects the design and operation of the plant. The SPAR model gives risk analysts the capability to quantify the expected risk of a NPP in terms of CDF and the change in that risk given an event, an anomalous condition, or a change in the design of the plant. More importantly, the model provides the analyst with the ability to identify and understand the attributes that significantly contribute to the risk and insights into how to manage that risk.

Currently, 78 SPAR models representing the 104 operating U.S. commercial NPPs are used for analysis of reactor core damage risk (Level 1 PRA) from internal events at-power. The Level 1 SPAR model includes an assessment of reactor core damage risk resulting from general transients (including anticipated transients without scram), transients induced by loss of a vital alternating current or direct current bus, transients induced by a loss of cooling (service) water, loss-of-coolant accidents, and loss of offsite power (LOOP). The SPAR models use a standard set of event trees for each plant design class and standardized input data for initiating event frequencies, equipment performance, and human performance, although these input data may be modified to be more plant- and event-specific, when needed. The system fault trees contained in the SPAR models are generally not as detailed as those contained in licensees' PRA models.

To more accurately model plant operation and configuration and to identify the significant differences between the licensee's PRA and SPAR logic, the staff performed detailed cut-set level reviews on all 78 models. In addition to the internal event at-power models, the staff developed 15 integrated external event models based on the licensee responses to the Individual Plant Examination of External Events (IPEEE) Program[32]; seven integrated low-power/shutdown models; and three extended Level 1 models supporting LERF and Level 2 modeling. The external event models were recently used to identify and evaluate severe accident sequences for the Consequential Steam Generator Tube Rupture Project in support of the NRC's Steam Generator Action Plan (ML003770259).

One significant upcoming activity is the incorporation into the SPAR models of internal fire

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[32] Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities, 10 CFR 50.54(f)" (November 23, 1988).

Supplement 4 to Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, 10 CFR 50.54(f)" (June 28, 1991).

scenarios from the NFPA 805 pilot applications. In addition, the staff continues to provide technical support for SPAR model users and risk-informed programs. The staff also completes about a dozen routine SPAR model updates annually.

The NRC implemented a formal SPAR model quality assurance plan in September 2006. Limited-scope validation and verification is accomplished by comparisons to licensee PRA models (as available) and to NRC NUREGs and analyses. Limited-scope peer reviews consist of internal quality assurance reviews by NRC contractors, NRC PRA staff, and regional SRAs (as available). Improvements to the models on a continuing basis result from staff user feedback, peer reviews from licensees, and insights gained from special studies, such as identification of threshold values during Mitigating Systems Performance Index (MSPI) reviews. In 2007, the NRC began a cooperative effort with the Electric Power Research Institute (EPRI) to improve PRA quality and address several key technical issues common to both the SPAR models and industry models. This cooperation resulted in the joint publication of a report that documents current methods to identify and quantify support system initiating events using PRAs[33]. Other cooperative projects include improvements to LOOP modeling and emergency core cooling system performance following boiling-water reactor (BWR) containment failure. In addition, the staff, with the cooperation of industry experts, performed a peer review of a representative BWR SPAR model and pressurized-water reactor (PWR) SPAR model in accordance with the industry consensus PRA standard for internal events, at-power Level 1/LERF PRAs[34] and RG 1.200[35]. The staff reviewed the peer review comments and initiated projects to address these comments where appropriate. The staff is also reevaluating certain success criteria in the SPAR models using state-of-the-art thermal-hydraulic modeling tools such as the MELCOR severe accident analysis code, which is described in more detail below.

<u>Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE), Version 8</u>

SAPHIRE 8 is a software application developed by the NRC for performing PRAs. SAPHIRE can be used to model a plant's response to initiating events, to quantify associated CDFs, and to identify important contributors to core damage (Level 1 PRA). It can also be used to evaluate containment failure and release models for severe accident conditions, given that core damage has occurred (Level 2 PRA). It can also be used in a limited manner to quantify risk in terms of release consequences to the public and environment (Level 3 PRA). It can be used for a PRA assuming that the reactor is operating at-power or in a low-power/shutdown POS. Furthermore, it can be used to analyze both internal and external initiating events, and it has special features for transforming models built for internal event analysis to models for external event analysis.

SAPHIRE 8 contains improved editors or options for creating event trees and fault trees, defining accident sequences and basic event failure data, solving system fault trees and accident sequence event trees, quantifying cut sets, performing sensitivity and uncertainty analyses, documenting the results, and generating reports.

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[33] EPRI Report 1016741, "Support System Initiating Events: Identification and Quantification Guideline" (2008).

[34] American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (February 2, 2009).

[35] Regulatory Guide 1.200, Rev. 2, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (March 2009).

SAPHIRE 8 is designed to easily handle larger and more complex models than previous versions. Applications of previous versions indicated the need to build and solve models with a large number of sequences. In addition, the complexity of the models has increased since PRAs evaluate both potential internal and external event initiators, as well as different POSs in which the initiating event may occur. Special features have been designed into SAPHIRE 8 to help create and run integrated models that may be composed of a number of different model types (e.g., models with different types of initiating events or POSs). External events models can be built more expeditiously through the use of automation tools. Any combination of model types can be solved, and a powerful graphical editor allows examination of the underlying logic.

New modeling and calculation methods have also been implemented. For example, phase mission time analysis capability was incorporated in support of the NRC's "extended Level 1" and limited-scope Level 2/LERF SPAR models; however, it may also be useful for low-power/shutdown models, which may consider multiple POSs. For CCF modeling, the Risk Assessment Standardization Project method has been incorporated, with CCF probabilities now automatically adjusted to account for the impact of sequence flag sets. In addition, SAPHIRE 8 offers an improved sequence solving algorithm which addresses limitations in the previous solving algorithm related to application of sequence recovery rules.

The uncertainty analysis functions in SAPHIRE 8 estimate the variability (due to the uncertainties in the basic event probabilities) of a fault tree top event probability, an event tree sequence frequency, and end state frequency, or any of the importance measures. In an uncertainty analysis, SAPHIRE 8 samples the user-specified distributions for each basic event in a group of cut sets, and then quantifies these cut sets using the sampled values.

One of the strengths of SAPHIRE 8 lies in its computational capabilities, which can easily be leveraged by non-expert users via an improved graphical user interface. SAPHIRE 8 has become a powerful and easy to use PRA tool. Its relational database structure and editing rules offer the capability for sophisticated modeling of accident progression and, therefore, offer the means for a more accurate and efficient analysis. Several other features, many constructed from feedback by users dealing with large-scale PRA models, make SAPHIRE 8 among the fastest and most sophisticated PRA codes available today.

MELCOR Severe Accident Analysis Code

The MELCOR severe accident analysis code is a fully integrated, engineering-level computer code whose primary purpose is to model the progression of postulated accidents in both LWRs and in non-reactor systems such as SFPs and dry storage casks.

MELCOR is a modular code consisting of three general types of packages:

(1) basic physical phenomena (e.g., hydrodynamics, heat and mass transfer to structures, gas combustion, aerosol and vapor physics),

(2) reactor-specific phenomena (e.g., decay heat generation, core degradation and relocation, ex-vessel phenomena, engineering safety systems), and

(3) support functions (e.g., thermodynamics, equations of state, material properties, data-handling utilities, equation solvers).

These packages model the major systems of a NPP and their associated interactions, including:

- Thermal-hydraulic response of the primary reactor coolant system, the reactor cavity, the containment, and the confinement buildings,
- Core uncovering (loss of coolant), fuel heatup, cladding oxidation, fuel degradation (loss of rod geometry), and core material melting and relocation,
- Heatup of reactor vessel lower head from relocated fuel materials and the thermal and mechanical loading and failure of the vessel lower head, and transfer of core materials to the reactor vessel cavity,
- Core-concrete attack and ensuing aerosol generation,
- In-vessel and ex-vessel hydrogen production, transport, and combustion,
- Fission product release (aerosol and vapor), transport, and deposition,
- Behavior of radioactive aerosols in the reactor containment building, including scrubbing in water pools, and aerosol mechanics in the containment atmosphere such as particle agglomeration and gravitational settling, and
- Impact of engineered safety features on thermal-hydraulic and radionuclide behavior

Initially, in the interest of quick code execution time and a general lack of understanding of reactor accident physics, the MELCOR code was envisioned as being predominantly parametric with respect to modeling complicated physical processes. However, over the years as phenomenological uncertainties have been reduced and user expectations and demands from MELCOR have increased, the models implemented into MELCOR have become increasingly best-estimate in nature. The increased speed and decreased cost of modern computers has eased many of the perceived constraints on MELCOR code development. Today, most MELCOR models are mechanistic, with capabilities approaching those of the most detailed codes of a few years ago. The use of models that are strictly parametric is limited, in general, to areas of high phenomenological uncertainty where there is no consensus concerning an acceptable mechanistic approach.

Current uses of MELCOR often include uncertainty analyses and sensitivity studies. To facilitate these uses, many of the mechanistic models have been coded with optional adjustable parameters. This does not affect the mechanistic nature of the modeling, but it does allow the analyst to easily address questions of how particular modeling parameters affect the course of a calculated transient. Parameters of this type, as well as such numerical parameters as convergence criteria and iteration limits, are coded in MELCOR as sensitivity coefficients, which may be modified through optional code input.

<u>MELCOR Accident Consequence Code System, Version 2 (MACCS2)</u>
MACCS2 represents a major enhancement of its predecessor MACCS, which was developed in 1990 to evaluate the impacts of severe accidents at NPPs on the surrounding public as part of the NUREG-1150 study. The principal phenomena considered are atmospheric transport and deposition under time-variant meteorology, short- and long-term mitigation actions and exposure pathways, deterministic and stochastic health effects, and economic costs. No other U.S. code that is publicly available at present offers all these capabilities.

MACCS2 was developed as a general-purpose tool applicable to diverse reactor and non-reactor facilities licensed by the NRC or operated by the Department of Energy or the Department of Defense. The MACCS2 package includes three primary enhancements (1) a more flexible emergency-response model, (2) an expanded library of radionuclides, and (3) a semi-dynamic food-chain model. Other improvements are in the areas of phenomenological modeling and new output options.

MACCS2 requires a substantial amount of supporting site-specific information, including, for example: meteorology, demography, land use, and property values. In addition, MACCS2 requires analysts make assumptions about the values of several parameters related to the implementation of mitigation actions following a severe accident. Examples include time needed to warn the public and initiate emergency response actions, effective evacuation speed, fraction of the offsite population that effectively participates in the emergency response actions, the degree of radiation shielding afforded by buildings, and projected dose limits. Uncertainty analyses would therefore be needed to understand the impact of parameter assumptions on the results.

An important limitation of MACCS2 is that it does not currently model and analyze the aqueous transport and dispersion of radioactive materials through surface water, sediments, soils, and groundwater. As demonstrated by the recent events in Japan, certain accident scenarios can result in large volumes of contaminated water being generated by emergency measures to cool the reactor cores and SFPs, with yet to be determined offsite radiological consequences. To determine the relative risk significance of these types of scenarios, a Level 3 PRA must be capable of modeling and analyzing the aqueous transport and dispersion of radioactive materials. This has therefore been identified as an important technical gap to be addressed as part of Option 2.

Site Selection Considerations

Although the objective of the Level 3 PRA scoping study was not to select a specific site for participation in a new full-scope comprehensive site Level 3 PRA, the staff identified a number of site selection considerations that can influence both the quality and availability of relevant information, as well as the resources needed to complete the study. These site selection considerations are presented below. Since licensee willingness to cooperate would be critical to success, it is important to recognize that the staff would have to engage with industry to identify and select the appropriate licensee for participation in the proposed study, should the Commission direct the staff to proceed with Option 3 or any other option requiring licensee cooperation.

Multi-Unit

Development of a full-scope comprehensive site Level 3 PRA model that can be used to understand the relative contribution of multi-unit effects to risk obviates the need for a multi-unit site.

SPAR Model Capability

Consistent with the proposed study objectives to enhance PRA capability and to demonstrate the technical feasibility and evaluate the cost of developing new Level 3 PRAs by leveraging both existing analytical tools and ongoing or planned relevant research, the staff would use an existing SPAR model as the starting point for developing the proposed full-scope comprehensive Level 3 PRA model. Sites with SPAR models that have been enhanced to incorporate external initiating event hazards, low-power/shutdown POSs, and/or Level 2 PRA technical elements are therefore good candidates for participation in the proposed study.

The potential enhancements that will be needed to the site-specific SPAR model will be driven primarily by what is needed to ensure the primary objective of obtaining new and improved risk insights is met. Due consideration will be given to requirements specified in industry consensus PRA standards and RG 1.200 to ensure technical adequacy of PRA results for risk-informed applications.

NFPA 805 Transition

To obtain credible results and insights from a fire PRA, a complete electronic cable raceway database and circuit analyses are needed. Development of these elements is extremely resource intensive and therefore cost prohibitive for the NRC. Sites participating in the voluntary transition to NFPA 805 implementation that have developed a state-of-the-art fire PRA that includes both of these elements are therefore good candidates for participation in the proposed study.

MELCOR Input Decks

Detailed MELCOR input decks that can support success criteria and accident progression calculations are both costly and time-consuming to develop. Sites participating in either the SOARCA project or in other ongoing research to investigate success criteria associated with specific Level 1 PRA sequences would already have detailed MELCOR input decks, and are therefore good candidates for participation in the proposed study.

Site-Specific External Hazards

The external initiating event hazards that are included in the scope of a PRA are determined by the site-specific hazards. The applicability of the insights that can be gained from one full-scope comprehensive site Level 3 PRA will depend in part on how representative the analyzed site is of the larger population of NPPs. Selecting a site with a representative set of external hazards may therefore be desirable. Alternatively, selecting a site with greater than normal external hazards of interest (e.g., seismic events, external flooding) can provide a different set of useful insights.

Spent Fuel Pool Storage Configuration

In attempting to understand the relative contribution of spent nuclear fuel to risk, another attribute to consider is the site-specific spent nuclear fuel storage configuration. For example, some sites use a common SFP for all reactors on the site, whereas others use separate SFPs for each reactor. Since the risk depends on the inventory of spent fuel that can be threatened, the site-specific storage configuration can have important risk implications.

Independent Spent Fuel Storage Installations (ISFSIs)

Sites with ISFSIs still have 4-5 cycles of spent nuclear fuel in their SFPs, which may be too hot to load into dry casks for storage. Because the risk associated with dry cask storage has been estimated to be lower than that for SFPs, this can have important risk implications. Performing the full-scope comprehensive site Level 3 PRA on a site that has an ISFSI can provide useful risk insights for other sites that also have ISFSIs. Alternatively, if a site without an ISFSI is selected, the PRA model can be used to obtain additional insights by assessing the risk significance of adding an ISFSI to the site.

The Structure and Evolution of Probabilistic Risk Assessment and Risk-Informed Regulation

PURPOSE

This document provides more detailed information on the structure and evolution of probabilistic risk assessment (PRA) and risk-informed regulation that led to the staff's original proposal for a scoping study to evaluate the feasibility of performing a new full-scope comprehensive site Level 3 PRA.

A separate document included as the first enclosure to the notation vote SECY paper provides more detailed technical information on (1) the basis for originally proposing to perform a scoping study to evaluate the feasibility of performing a new full-scope comprehensive site Level 3 PRA for a nuclear power plant (NPP)[1], (2) potential future uses for Level 3 PRAs, (3) three primary options for proceeding with future Level 3 PRA activities[2], and (4) the activities that supported development of items 2 and 3.

[1] As used in this document and the SECY paper to which it is enclosed, a full-scope comprehensive site Level 3 PRA is a PRA that includes a quantitative assessment of the public risk from accidents involving all site reactor cores and spent nuclear fuel that can occur during any plant operating state, and that are caused by all initiating event hazards (internal events, fires, flooding, seismic events, and other site-specific external hazards).

[2] This document and the SECY paper to which it is enclosed distinguish between "Level 3 PRA activities" and "Level 3 PRAs." The latter refers to a PRA that includes specific technical elements or analyses to assess the public risk from a NPP, while the former refers to activities (e.g., research and development) specifically related to or in support of Level 3 PRAs.

The Structure and Evolution of Probabilistic Risk Assessment and Risk-Informed Regulation

BACKGROUND

A Quantitative Definition of Risk

The traditional definition of risk involves the combination of the likelihood of and consequences associated with an adverse event. Kaplan and Garrick[3] advanced this definition and formalized risk as a set of triplets developed by answering the following three questions:

(1) What can go wrong?
(2) How likely is it that it will happen?
(3) If it does happen, what are the consequences?

To answer these questions, a set of possible scenarios or outcomes are identified, each with an associated probability and consequence measure. The total risk (R) is therefore captured by the set of all possible scenarios identified (s), the probabilities of those scenarios occurring (p), and the consequence measures of those scenarios (x). In equation form,

$$\mathbf{R = \{(s_i, p_i, x_i)\}, \qquad i = 1, 2, \ldots, N}$$

Probabilistic Risk Assessment (PRA)

PRA is a structured, analytical process that provides both qualitative insights and quantitative estimates of risk by (1) identifying potential sequences that can challenge system operations and lead to an adverse event, (2) estimating the likelihood of these sequences, and (3) estimating the consequences associated with these sequences, if they were to occur. By prioritizing significant risk contributors[4] and characterizing key sources of uncertainty and their impact on results, PRA serves as a useful decisionmaking tool that can help focus thinking and limited agency resources to ensure safety.

The Use of PRA in the Decisionmaking Process

In using PRA as a tool to support a regulatory decision, the following four-step process is typically followed:

(1) **Identify the results needed**. For many risk-informed applications, acceptance criteria or guidelines have been established in terms of numerical values of risk metrics. Therefore, when using PRA results to support a risk-informed decision, the first step is to identify which results are needed and how they are to be used to inform the decision.

(2) **Construct a PRA model to generate the required results**. Once identified, the next step is to develop a model, typically a quantitative PRA model that is of the appropriate scope and level of detail that can generate the needed results.

(3) **Compare PRA results to acceptance criteria or guidelines**. Once results are generated, they can be compared to the appropriate acceptance criteria or guidelines. Although this appears to be straightforward, this step involves more than

[3] Kaplan S. and Garrick B.J., "On the quantitative definition of risk." Risk Analysis, 1, 11-37 (1981).

[4] As used in this enclosure and the SECY paper to which it is enclosed, risk contributors include: radiological sources (e.g., reactor core, spent nuclear fuel); initiating event hazards (e.g. internal events, fires, flooding, seismic events, other site-specific external hazards); plant operating states; accident sequences; failure of structures, systems, and components; and operator actions.

just a simple comparison of numerical values. To ensure confidence in the decision, the PRA results need to be evaluated to determine their realism and to identify and address any key sources of uncertainty. Types and sources of uncertainty in PRA models and results are discussed in more detail below.

(4) **Document the results**. In this final step, the results of the comparison of the PRA results to the acceptance criteria or guidelines are documented, along with a statement characterizing the confidence in the results.

Characteristics of NPP PRA Models

To understand why future Level 3 PRAs would be beneficial, it is important to first understand some of the key characteristics of NPP PRA models that can influence their use in regulatory applications, including the scope, level of detail, structure, associated uncertainties, and the aggregration of PRA results from different hazards.

<u>The Scope of a PRA Model</u>

NPP PRA models can vary in scope, depending on their intended application or use. As summarized in Table 1 below, the scope of a PRA is defined by the extent which various options for the following five factors are modeled and analyzed:

(1) Radiological sources. NPP sites contain multiple sources that could potentially release radioactive material into the environment under accident conditions. Although current PRAs focus on the reactor core, other important sources that could be modeled in the PRA to estimate the public risk from NPP sites include spent nuclear fuel (both wet and dry storage), fresh fuel, and radiological waste storage tanks.

(2) Population exposed to the hazards. In determining the potential health effects associated with a nuclear accident, both onsite and offsite populations can be considered. Typical NPP PRA models have been developed to estimate the risk to the general public located offsite, and do not consider the risk to the onsite workers and immediate responders to a nuclear accident.

(3) Initiating event hazard groups. Initiating events disrupt the steady state operation of the plant by challenging plant control and safety systems and operators whose failure could potentially lead to reactor core damage and/or the release of radioactive material to the environment. These events include failure of equipment from internal causes (e.g., transients, loss-of-coolant accidents, internal floods, internal fires) or external causes (e.g., earthquakes, high winds, tsunamis). In a NPP PRA model, similar causes of initiating events are organized by hazard groups, which are then assessed using common approaches, methods, and data to characterize their effects on the plant.

(4) Plant operating states (POSs). In determining the public risk from NPP operations, it is important to consider not only the response of the plant to initiating events occurring during at-power operation, but also its response to initiating events occurring while the plant is in other operating states, such as low-power and shutdown (LPSD). POSs are used to subdivide the plant operating cycle into unique states defined by various characteristics (e.g., reactor power; coolant temperature, pressure, and level; equipment configuration) so that the plant response can be assumed to be the same for all subsequent initiating events.

(5) End state (level of risk characterization). NPP PRA models can be used to calculate risk metrics at different end states. The three different end states or levels of risk characterization that have been traditionally used in NPP PRA models are discussed in more detail below.

Table 1. Scoping Options for Commercial NPP PRAs

Factor	Scoping Options for Commercial NPP PRAs
Radiological sources	Reactor core(s) Spent nuclear fuel (spent fuel pool and dry cask storage) Other radioactive sources (e.g., fresh fuel and radiological wastes)
Population exposed to hazards	Onsite population Offsite population
Initiating event hazard groups	Internal hazards • Traditional internal events (transients, loss-of-coolant accidents) • Internal floods • Internal fires
	External hazards • Seismic events (earthquakes) • Other site-specific external hazards (e.g., high winds, external flooding)
Plant operating states	At-Power Low-Power/Shutdown
End state/Level of risk characterization	Level 1 PRA: Initiating event to onset of core damage or safe state Level 2 PRA: Initiating event to radioactive material release from containment Level 3 PRA: Initiating event to offsite radiological consequences

When using PRA to support regulatory applications, all risk contributors relevant to the regulatory decision need to be included in the scope of the PRA model. In accordance with staff requirements memorandum (SRM) COMNJD-03-0002[5], the risk from each significant risk contributor should be addressed using a PRA model developed in accordance with a U.S. Nuclear Regulatory Commission (NRC) staff-endorsed consensus standard. In some cases, however, a conservative bounding assessment or qualitative screening analysis can be used to demonstrate that some risk contributors are not relevant to the regulatory decision, and can therefore be excluded from the scope of the PRA.

Level of Detail of a PRA Model

Much like scope, the level of detail of a NPP PRA model can vary, depending on its intended application or use. The level of detail is defined by the degree to which (1) the actual plant is modeled and (2) the unlimited range of potential scenarios is simplified. Although the goal of a PRA is to represent the as-designed, as-built, and/or as-operated plant as realistically as practicable, some compromise must be made to keep the PRA model manageable.

For each of the technical elements that comprise a PRA model, the level of detail may vary by the extent to which:

[5] SRM COMNJD-03-0002, "Stabilizing the PRA Quality Expectations and Requirements" (September 8, 2003).

(1) Plant systems and operator actions are credited in modeling plant-specific design and operation,

(2) Plant-specific operating experience and data for the plant's structures, systems, and components (SSCs) are incorporated into the model, and

(3) Realism is incorporated into analyses that predict the expected plant and operator responses.

In addition, to keep the PRA model manageable, the logic structures (e.g., event trees and fault trees) used in the model are simplified representations of the complete range of potential accident scenarios. Simplifications are made through underlying assumptions and approximations, such as the consolidation of initiating event causes into representative hazard groups and the screening out of certain equipment failure modes.

Although the level of detail needed for a NPP PRA model is largely dependent upon the requirements associated with its intended use, at a minimum, the model needs to be detailed enough to model the major system dependencies and to capture the significant risk contributors.

The Structure of a PRA Model

NPP PRA models are logic models constructed using logic structures such as event trees and fault trees. Event trees are used to model different plant and operator responses in terms of sequences of undesired system states that could occur following an initiating event. Fault trees are used to identify different combinations of basic events (e.g., initiating events, SSC failures, and human failure events) that could lead to the undesired system states. When linked together, these logic structures provide an integrated perspective that can capture major system dependencies.

As discussed in the previous section, these logic structures represent a simplification of the potentially unlimited range of scenarios by modeling a more manageable yet representative set that encompasses all of the potential consequences. Underlying assumptions and approximations made in the development of the PRA model give rise to uncertainty, a topic discussed in more detail below.

Uncertainties in PRA Models

When using PRA results as part of any regulatory decisionmaking process, it is important to understand the types, sources, and potential impact of uncertainties associated with PRA models and how to treat them in the decisionmaking process. NUREG-1855[6] was developed to address these issues.

Although there are several different sources of uncertainty in PRA models, there are two principal classes of uncertainty: aleatory and epistemic. Aleatory uncertainty arises from the random nature of the basic events modeled in PRAs. Because PRAs use probabilistic distributions to estimate the frequencies or probabilities of these basic events, the PRA model itself is an explicit model of the aleatory uncertainty.

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[6] NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (March 2009).

Epistemic uncertainties arise from incompleteness in the collective state of knowledge about how to represent plant behavior in PRA models. These uncertainties relate to how well the PRA model reflects the as-designed, as-built, and/or as-operated plant and to how well it predicts the response of the plant to various scenarios. Since these uncertainties can have a significant impact on the interpretation and use of PRA results, it is important that they be appropriately identified, characterized, and addressed. The three types of epistemic uncertainty associated with PRA models are:

(1) **Parameter Uncertainty.** Parameter uncertainty relates to uncertainty in the computation of input parameters for the probability distributions used to quantify the frequencies or probabilities of basic events in the PRA logic model. Importantly, this assumes that the selection of the probability distribution used to model the likelihood of the basic event is agreed upon; if uncertainty exists about this selection, it is more appropriately considered model uncertainty. Parameter uncertainty is typically characterized by using probability distributions to represent the degree of belief in the values of these input parameters.

(2) **Model Uncertainty.** Model uncertainty arises from a lack of knowledge of physical phenomena, failure modes related to the behavior of SSCs under various conditions, or other phenomena modeled in a PRA (e.g., the location and habits of members of the public in different exposure scenarios). This can result in the use of different approaches to modeling certain aspects of the plant and public response that can significantly impact the overall PRA model. Since uncertainty exists about which approach is most appropriate, this leads to uncertainty in the PRA results. Model uncertainty can also arise from uncertainty in the logic structure of the PRA model or in the selection of the probability distribution used to model the likelihood of the basic events in the PRA model. Model uncertainties are typically addressed by using sensitivity analyses to determine the sensitivity of the PRA results to any reasonable alternative modeling approaches.

(3) **Completeness Uncertainty.** Completeness uncertainty arises from limitations in the scope and completeness of the PRA model. Known risk contributors can be excluded from the PRA model due to technology or resource limitations or because their contribution to overall risk is believed to be negligible. These uncertainties can be addressed by supplementing the PRA with additional analyses to demonstrate their impact is not significant. Unknown risk contributors are excluded because their potential existence has not yet been recognized. These uncertainties are typically addressed through the use of defense-in-depth principles. Although it can be viewed as a special type of model uncertainty, completeness uncertainty is treated separately because it reflects an unanalyzed contribution to risk that is difficult, if not impossible, to quantify.

Skeptics of PRA question its usefulness due to the uncertainty in its results. Although PRA cannot account for the unknown and identify all unexpected event scenarios, it can identify some originally unforeseen scenarios, identify where some of the uncertainties exist in plant design and operation, and for some uncertainties, quantify the extent of the uncertainty. PRA is therefore a powerful tool that can lead to safer plant design by focusing attention and resources on those aspects important to safety and by identifying where defense-in-depth measures are needed to account for uncertainty.

The Aggregation of PRA Results from Different Hazards
PRA results can include more than just calculated risk metrics for comparison to acceptance criteria or guidelines. In fact, one of the most valuable insights from PRA can be the identification of the relative importance of various risk contributors.

For many regulatory applications, it is necessary to consider the contributions from several hazards to a specific risk metric. When considering multiple hazards, a PRA model can be a fully integrated model in which all hazards are combined into a single logic structure, a set of individual PRA models for each hazard, or a mixture of the two. When combining the results of PRA models for several hazards, the level of detail and level of approximation included in the PRA model may differ from one hazard to the next. Because of the unique methods and data used, a significantly higher level of conservative bias can exist in PRAs for internal fires, external events (seismic, high wind, and others), and low-power/shutdown conditions. In principal, this conservative bias could be reduced to some degree by developing models to the same level of detail and rigor associated with internal events, at-power PRAs. That said, the larger conservative bias can result in larger uncertainties in the results. Importantly, however, this does not preclude the aggregation of results from different hazards. Instead, it requires an understanding of the main sources of conservatism associated with any of the hazards that can potentially impact the regulatory application for which the PRA results are being used.

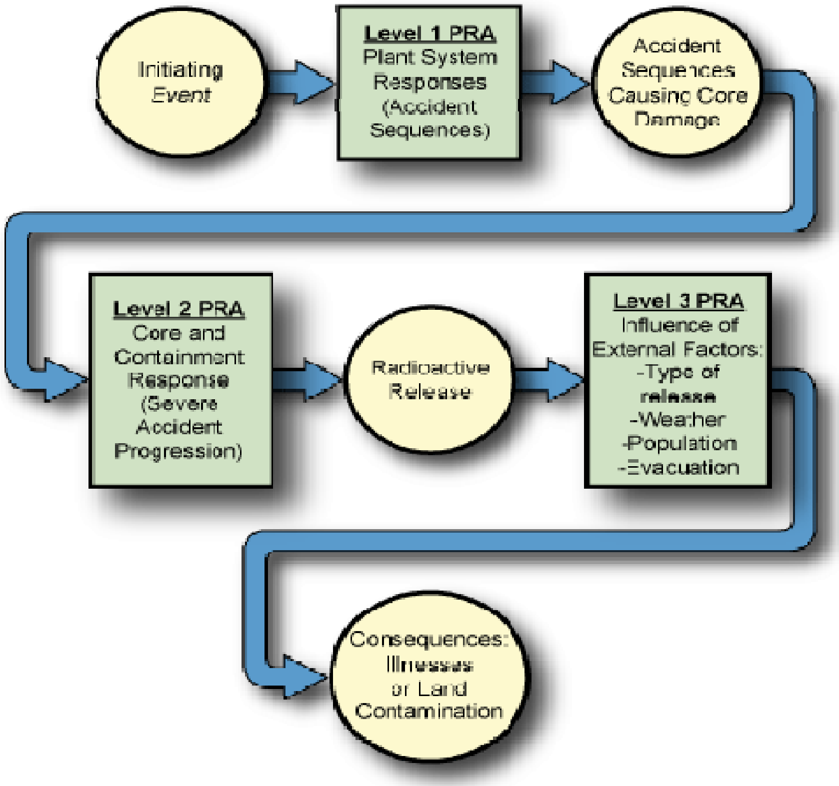
Therefore, when interpreting the results of the comparison of risk metrics to acceptance criteria or guidelines, it is important to not only focus on the aggregated numerical result, but also on the relative importance and realism of the main contributors to the risk metric.

PRA End States: The Significance of Level 3 PRAs
As shown in Table 1 and Figure 1 below, NPP PRAs that have traditionally focused on accidents involving the reactor core can estimate risk metrics at three different levels of characterization by using sequential analyses in which the output from one level serves as a conditional input to the next.

Level 1 PRA
Using event trees and fault trees, a Level 1 PRA models system and operator responses to various initiating events that challenge plant operation to identify sequences (combinations of system and operator action successes and failures) that result in either the achievement of a safe state or the onset of core damage. The estimated frequencies of those sequences that result in the onset of reactor core damage are summed to calculate the total core damage frequency (CDF) for the analyzed plant.

Level 2 PRA
A Level 2 PRA includes Level 1 PRA analyses and in addition estimates conditional containment failure probabilities (CCFPs), radioactive material release frequencies, and various source term characteristics by modeling the progression of those accident sequences resulting in core damage (otherwise known as “severe accidents”) and evaluating the response of both plant systems and the containment to the harsh accident environment. For those sequences resulting in containment failure or bypass, the frequency, type, amount, timing, and energy content of the radioactive material released to the environment is estimated.

Figure 1. PRA End States: The three sequential levels of risk characterization.



<u>Level 3 PRA</u>
A Level 3 PRA includes Level 2 PRA analyses and in addition models atmospheric transport and dispersion phenomena to estimate various offsite health and economic consequence measures, Inputs to the Level 3 PRA include the source term characteristics from a Level 2 PRA and several other factors, to include site-specific meteorology, demographics, emergency response, and land use. Outputs from the Level 3 PRA include estimates of offsite radiological consequences in terms of various consequence measures such as early fatalities and injuries and latent cancer fatalities resulting from the radiation doses to the surrounding population, and economic costs associated with evacuation, relocation, property loss, and land contamination.

Importantly, by combining the radioactive material release frequencies obtained from a Level 2 PRA with the offsite radiological consequences associated with each release, only a Level 3 PRA can estimate the public risk from all analyzed risk contributors associated with a NPP. More importantly, a full-scope comprehensive site Level 3 PRA that includes an assessment of not only accidents involving the reactor core, but also accidents involving other site radiological sources—such as spent fuel pools (SFPs), dry storage casks, and other units[7] on site–can provide valuable insights into the relative importance of various site risk contributors. These insights can be used to enhance regulatory decisionmaking and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety. Although a PRA that includes an assessment of other site radiological sources could conceivably be done using only a Level 1 or Level 2 PRA, such an assessment would not

[7] As used in this enclosure and the SECY paper to which it is enclosed, a unit refers to a reactor core and, if applicable, an associated spent fuel pool.

necessarily yield information about issues most directly related to the agency's mission to protect public health and safety.

A common misconception is that a Level 2 PRA is limited to the accident progression and source term analyses, while a Level 3 PRA is limited to the accident consequence analyses. To be clear, a Level 2 PRA includes the analyses from a Level 1 PRA, and a Level 3 PRA includes the analyses from a Level 2 PRA. A Level 3 PRA therefore analyzes from initiating event to offsite radiological consequences for accident sequences involving core damage and containment failure or bypass. The analyses that are included in each PRA level are summarized below in Table 2.

Table 2. Analyses Included in Each PRA Level

Analysis	Level 1 PRA	Level 2 PRA	Level 3 PRA
Accident Frequency Analysis	X	X	X
Accident Progression Analysis		X	X
Source Term Analysis		X	X
Consequence Analysis			X
Risk Integration Analysis			X

Historical Perspective: The Evolution of PRA Technology and Risk-Informed Regulation

Pre-PRA Policy Statement Era (1946 – 1995)

From 1946 to 1954, nuclear regulation was the responsibility of the NRC's predecessor, the Atomic Energy Commission (AEC). The Atomic Energy Act of 1946 established the AEC to maintain strict control of atomic technology and to further exploit it for military applications. By 1954, the need for commercial nuclear power became an urgent national goal, and a new Atomic Energy Act was passed. Under this Act, the AEC had responsibility for the development and production of nuclear weapons and for both the development and the safety regulation of the civilian uses of nuclear materials.

In the development of early nuclear safety regulations, the AEC ensured adequate protection of public health and safety by using a conservative deterministic approach to demonstrate that NPPs could withstand a set of worst-case design basis accidents (DBAs) involving single failures in independent systems following certain initiating events. In addition, the AEC relied on the concept of defense-in-depth, which originated in the design of nuclear weapons facilities to account for uncertainties in safety system design margins. This concept, which promotes the use of safety margins and multiple, independent layers of defense mechanisms, would theoretically mitigate the consequences of a severe accident resulting in core damage, or "beyond-DBA," should one occur.

Prior NRC-Sponsored Studies to Estimate Public risk

As NPP designs and PRA techniques evolved over time, the NRC and its predecessor, the AEC, periodically sponsored studies to obtain updated estimates of the public risk from severe reactor accidents.

WASH-740[8]
Published in March 1957 by the AEC, the purpose of this first major study was to provide an estimate of the upper limit consequences of severe reactor accidents to inform Congressional deliberation on the Price-Anderson Act. The study was conservative in nature, focusing on large loss-of-coolant accidents (LOCAs) as the leading source of worst-case radioactive material release to the environment. Although this was a non-probabilistic consequence study instead of PRA study, the scientists involved were willing to offer rough order-of-magnitude estimates of the probability of a severe reactor accident that ranged from 10^{-5} – 10^{-9} per reactor-year of operation.

WASH-1400[9]
During the late 1960s and early 1970s, the size and number of commercial NPPs rapidly increased. In addition, a series of loss-of-fluid tests (LOFTs) conducted using a small-scale reactor mockup suggested that steam buildup during an accident scenario could prevent the Emergency Core Cooling System (ECCS) from injecting water into the reactor core, thereby leading to core damage. In the midst of these concerns with ECCS performance and an upcoming extension of the Price-Anderson Act, the AEC initiated a study in 1972 to obtain a more realistic estimate of the public risk from severe nuclear accidents.

In October 1975,18 years after the publication of WASH-740, and after considerable progress in the use of reliability techniques and increased use of commercial NPPs, the NRC published WASH-1400. This Level 3 PRA study marked the first U.S. attempt to systematically evaluate a large spectrum of accidents and to use quantitative techniques to evaluate severe accident probabilities, source terms, and offsite radiological consequences in an integrated manner to obtain a more realistic estimate of severe accident public risk.

The WASH-1400 study demonstrated that although the CDF and the CCFP, given the occurrence of accident sequence that releases radioactive material into the containment atmosphere, were both higher than previously estimated, the offsite radiological consequences associated with these severe reactor accidents were much smaller.

More important than the actual risk estimates were the risk insights that were gained. The WASH-1400 study challenged the concept that conservative safety analyses of DBAs could establish an upper limit on public risk. Small-break LOCAs and other accident sequences involving multiple failures were found to contribute much more significantly to risk than the large-break LOCA DBAs involving single failures.

Although the PRA methodology used in WASH-1400 was broadly endorsed as the best available at the time, the study was widely criticized for its treatment of uncertainties in its estimates of severe accident probabilities. In fact, in January 1979, the Commission withdrew its support of the WASH-1400 results stating, “In particular, in light of the [Risk Assessment] Review Group conclusions on accident probabilities, the Commission does not regard as reliable the Reactor Safety Study’s estimate of the overall risk of a reactor accident.” Three months after the Commission released this statement, the accident at Three Mile Island (TMI)

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[8] WASH-740, “Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants” (March 1957).

[9] WASH-1400, “Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants” (October 1975).

occurred. This seminal event, which substantiated the WASH-1400 insight that small-break LOCAs were more significant contributors to risk than large-break LOCA DBAs, led to the initiation of a substantial research program on severe accident phenomenology and the increased use of PRA to identify plant vulnerabilities in the nuclear industry.

NUREG-1150[10]

As part of its integration plan for closure of severe accident issues[11], the NRC staff initiated a follow-on Level 3 PRA study in 1986 to update WASH-1400 using advanced PRA technology that could include quantitative estimates of risk uncertainty. Published in December 1990, 15 years after WASH-1400, the NUREG-1150 study provided a set of PRA models and a snapshot-in-time assessment of the severe accident risks associated with five commercial NPPs of different reactor and containment designs. The reactor and containment design for each of the sites involved, as well as the scope of initiating event hazard groups analyzed in each PRA are summarized below in Table 3. These "full-scope" PRAs were limited to the assessment of single-unit reactor accidents initiated primarily by internal events occurring during at-power operations, with only a partial treatment of fires and seismic events for two of the five analyzed plants. A later study evaluated the risk associated with accident sequences occurring during low-power/shutdown operations for two of the five analyzed plants (Grand Gulf[12] and Surry[13]).

Table 3. NUREG-1150 Reactor/Containment Design and Initiating Event Hazard Groups

Reactor/Containment Design	Level 1 PRA Scope	Level 2 PRA Scope	Level 3 PRA Scope
Surry-1 • Westinghouse 3-loop • Subatmospheric	Internal Events Fires Seismic Events	Internal Events Fires Seismic Events	Internal Events Fires
Zion-1 • Westinghouse 4-loop • Large dry	Internal Events	Internal Events	Internal Events
Sequoyah-1 • Westinghouse 4-loop • Ice condenser	Internal Events	Internal Events	Internal Events
Peach Bottom-2 • BWR-4 • Mark I	Internal Events Fires Seismic Events	Internal Events Fires Seismic Events	Internal Events Fires
Grand Gulf-1 • BWR-6 • Mark III	Internal Events	Internal Events	Internal Events

Primarily through the use of improved data and sophisticated models, the NUREG-1150 PRAs showed that estimates of severe accident risks were even lower than those provided by the WASH-1400 PRAs. More importantly, as a landmark study that advanced the state-of-the-art in

[10] NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" (December 1990).

[11] SECY-88-147, "Integration Plan for Closure of Severe Accident Issues" (May 25, 1988).

[12] NUREG/CR-6143, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Grand Gulf, Unit 1" (March 1995).

[13] NUREG/CR-6144, "Evaluation of Potential Severe Accidents During Low Power and Shutdown Operations at Surry, Unit 1" (July 1994).

PRA, particularly in terms of the uncertainty analysis, the NUREG-1150 models, results, and risk perspectives would subsequently be used in a variety of regulatory applications including, but not limited to:

- Development and implementation of the PRA Policy Statement
- Validation of regulatory analysis guidelines
- Validation of subsidiary numerical objectives
- Support for risk-informed rulemaking
- Prioritization of generic safety issues and nuclear safety research programs
- Individual Plant Examination of External Events (IPEEE) Program

Safety Goal Policy Statement[14]

In 1986, still in the aftermath of the TMI accident, the Commission issued the Safety Goal Policy Statement, in which it broadly defined an acceptable level of public risk due to NPP operations by establishing two qualitative safety goals, each supported by an associated quantitative health objective (QHO). These safety goals and their supporting QHOs are summarized below in Table 4.

Table 4. The Commission's Safety Goals for the Operations of NPPs

Qualitative Safety Goal	Associated QHO
Individual members of the public should be provided a level of protection from the consequences of NPP operation such that individuals bear no significant additional risk to life and health.	The risk to an average individual in the vicinity of a NPP of prompt fatalities that might result from reactor accidents should not exceed 0.1% of the sum of prompt fatality risks resulting from other accidents to which the members of the U.S. population are generally exposed.
Societal risks to life and health from NPP operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.	The risk to the population in the area near a NPP of cancer fatalities that might result from NPP operation should not exceed 0.1% of the sum of cancer fatality risks resulting from all other causes.

In its guidelines for regulatory implementation, the Commission directed the staff to develop specific guidance for use as a basis for determining whether a level of safety ascribed to a plant is consistent with the safety goal policy. The Commission indicated that this guidance would be based on the following general performance guideline proposed for further staff examination:

> *"Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation."*

In response, the NRC staff proposed that the safety goals and QHOs be partitioned into further subsidiary objectives that could utilize the risk metrics from Level 1, Level 2, and Level 3 PRAs as a basis for comparison. Although the Commission rejected this proposal in the associated

[14] 51 FR 30028, "Safety Goals for the Operations of Nuclear Power Plants" (August 21, 1986).

SRM[15], it supported the use of subsidiary quantitative core damage frequency and containment performance objectives through partitioning of the proposed large release guideline. Consistent with the defense-in-depth philosophy, these subsidiary objectives could be used as minimum acceptance criteria for prevention (core damage frequency) and mitigation (containment performance).

Direct comparison with the existing QHOs requires a Level 3 PRA that estimates the risk from all analyzed risk contributors associated with NPP operations. However, when progressing from determining the frequencies of accident sequences to estimating offsite radiological consequences, the calculations become more complex and costly, with increasing uncertainty in the end results. With Commission support, the staff therefore utilized NUREG-1150 results to develop and adopt the following subsidiary numerical objectives that could be compared with the results of Level 1 and Limited Level 2 PRAs:

- Core damage frequency (CDF) < 10^{-4} per reactor-year (surrogate for cancer fatality QHO)
- Large early-release frequency (LERF) <10^{-5} per reactor-year (surrogate for prompt fatality QHO)

The development of these subsidiary numerical objectives played an important role in the implementation of risk-informed regulation, and is germane to some of the issues that exist within the current risk-informed regulatory framework.

<u>Individual Plant Examination (IPE) Program</u>
On August 8, 1985, the Commission issued its policy statement on "Severe Reactor Accidents Regarding Future Designs and Existing Plants" (50 FR 32138), which introduced the Commission's plan to address severe accident issues for existing commercial NPPs. During the next few years, the Commission formulated an approach for systematically evaluating the safety of NPPs to identify particular accident vulnerabilities and cost-effective changes to ensure no undue risk to public health and safety.

To implement this approach, the NRC issued Generic Letter (GL) 88-20[16], requesting that each licensee perform a plant examination to "*identify plant-specific vulnerabilities to severe accidents that could be fixed with low-cost improvements*." The specific objectives of the IPE program were for each utility to:

(1) Develop an overall appreciation of severe accident behavior;

(2) Understand the most likely severe accident sequences that could occur at its plant;

(3) Gain a more quantitative understanding of the overall probability of core damage and radioactive material releases; and

(4) If necessary, reduce the overall probability of core damage and radioactive material release by appropriate modifications to procedures and hardware that would help prevent or mitigate severe accidents.

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[15] SRM SECY-89-102, "Implementation of the Safety Goals" (June 15, 1990).

[16] Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities, 10 CFR 50.54(f)" (November 23, 1988).

In GL 88-20, the NRC identified PRA as one acceptable approach for conducting an IPE and further identified a number of potential benefits associated with performing PRAs on those plants without one. Examples of benefits included (1) support for licensing actions, (2) license renewal, (3) risk management, and (4) integrated safety assessment. As a result, licensees elected to perform PRAs for their IPEs.

The NRC staff received and evaluated 75 IPE submittal PRAs covering 108 NPP units. Based on guidance provided in GL 88-20, the scope of these Level 1 and Level 2 PRAs was limited to internal initiating events (including internal flooding) occurring during at-power operations. Even with these scope limitations, the NRC staff concluded that licensees had generally developed internal capability with an increased understanding of PRA and severe accidents and that the IPE Program had served as a catalyst for further improving NPP safety. Perspectives gained from the IPE Program are summarized in NUREG-1560[17].

Individual Plant Examination of External Events (IPEEE) Program

In June 1991, the NRC issued Supplement 4 to GL 88-20[18], requesting that "each licensee perform an individual plant examination of external events to identify vulnerabilities, if any, to severe accidents and report the results together with any licensee-determined improvements and corrective actions to the Commission." The external events considered in the IPEEE program included: internal fires; seismic events; and high winds, floods, and other (HFO) external initiating events involving accidents related to transportation and nearby facilities. Deliberate malevolent acts (e.g., sabotage, terrorism) were not included in the set of events considered. The objectives of the IPEEE Program were consistent with those of the IPE Program.

The NRC staff received and evaluated 70 IPEEE submittal PRAs covering all operating U.S. NPPs at the time. Through its review and evaluation, the NRC staff concluded that the perspectives and insights gained from the IPEEE program would be particularly useful in (1) NRC and industry risk-informed regulatory initiatives and activities, (2) guidance for future external events standards and PRAs, and (3) prioritization of research to improve risk analysis methods. Perspectives gained from the IPEEE program are summarized in NUREG-1742[19].

Post-PRA Policy Statement Era (1995 – Present)

As PRA technology matured and as confidence in the nuclear industry's use of PRA to positively impact NPP safety increased through the IPE Program, the NRC gradually refined its deterministic regulatory framework by incorporating the use of risk information and insights in a risk-informed regulatory framework. In 1994, the NRC developed the PRA Implementation Plan[20] to focus its efforts on increasing the use of PRA in regulatory activities. This plan was

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[17] NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance" (December 1997).

[18] Supplement 4 to Generic Letter 88-20, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities, 10 CFR 50.54(f)" (June 28, 1991).

[19] NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program" (April 2002).

[20] SECY-94-219, "Proposed Agency-wide Implementation Plan for Probabilistic Risk Assessment" (August 19, 1994).

superseded in 2000 by the Risk-Informed Regulation Implementation Plan (RIRIP)[21], which was developed to more clearly describe the NRC's risk-informed activities and to provide links between those activities and the NRC's Strategic Plan. Finally, in April 2007, the NRC replaced the RIRIP with the Risk-Informed, Performance-Based Plan (RPP)[22], an integrated master plan for initiatives designed to help the NRC achieve the Commission's goal of a holistic, risk-informed and performance-based regulatory framework. Each of these plans has guided the NRC in developing risk-informed, performance-based regulations.

In this section, some of the more important activities that have shaped the development and implementation of the existing risk-informed regulatory framework are highlighted. In addition, to further set the stage for providing a basis for proposing new Level 3 PRA activities, an overview of how risk-information is currently used in regulatory activities is provided.

PRA Policy Statement[23]

On August 16, 1995, the Commission issued its PRA Policy Statement, which effectively introduced the risk-informed regulatory paradigm. Established to promote regulatory stability and efficiency through consistent and predictable implementation of potential PRA applications, this Commission policy included the following four main statements:

(1) The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art and in a manner that complements the NRC's deterministic approach and traditional defense-in-depth philosophy.

(2) Where practical within the bounds of the state-of-the-art, PRA should be used to reduce unnecessary conservatism in current regulatory requirements and to support proposals for additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule).

(3) PRAs used in support of regulatory decisions should be as realistic as practicable.

(4) The Commission's safety goals and subsidiary numerical objectives are to be used with appropriate consideration of uncertainties in making regulatory decisions.

Regulatory Guide (RG) 1.174[24]

Although it was developed to address the use of PRA in only a specific subset of the applications identified in the PRA Implementation Plan, RG 1.174 establishes a framework for risk-informed integrated decisionmaking that has been generalized to apply to a wide variety of applications, including other application-specific regulatory guides developed to risk-inform inservice testing[25], technical specifications[26], and inservice inspection of piping[27].

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[21] SECY-00-0062, "Risk-Informed Regulation Implementation Plan" (March 15, 2000).

[22] SECY-07-0191, "Implementation and Update of the Risk-Informed and Performance-Based Plan" (October 31, 2007).

[23] 60 FR 42622, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities" (August 16, 1995).

[24] Regulatory Guide 1.174, Rev. 2, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (May 2011).

[25] Regulatory Guide 1.175, "An Approach for Plant-Specific Risk-Informed Decisionmaking: Inservice Testing" (August 1998).

This risk-informed integrated decisionmaking framework, which consists of five key principles, was developed to improve consistency in regulatory decisions where PRA results are used to supplement traditional deterministic and defense-in-depth approaches. The five key principles include:

(1) **Current Regulations Met.** The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.

(2) **Consistent with Defense-in Depth.** The proposed change is consistent with the defense-in-depth philosophy.

(3) **Maintains Safety Margins.** The proposed change maintains sufficient safety margins.

(4) **Acceptable Risk Impact.** When proposed changes result in an increase in CDF or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

(5) **Monitor Performance.** The impact of the proposed change should be monitored using performance measurement strategies.

Principle 4 relates specifically to the use of PRA results. For the purposes of RG 1.174, the proposed change is considered to have met the intent of the Commission's Safety Goal Policy Statement if the PRA results meet established acceptance guidelines based on a comparison of the change in CDF and LERF to the total baseline CDF and LERF, respectively. Taken from RG 1.174, figures 2 and 3 below illustrate the CDF and LERF acceptance guidelines, respectively.

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[26] Regulatory Guide 1.177, Rev. 1, "An Approach for Plant-Specific Risk-Informed Decisionmaking: Technical Specifications" (May 2011).

[27] Regulatory Guide 1.178, Rev. 1, "An Approach for Plant-Specific Risk-Informed Decisionmaking for Inservice Inspection of Piping" (September 2003).

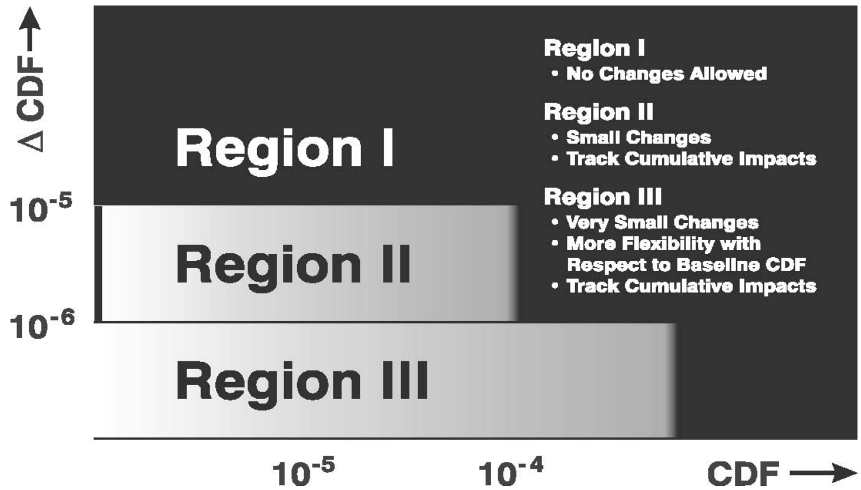

Figure 2. RG 1.174 Acceptance Guidelines for CDF

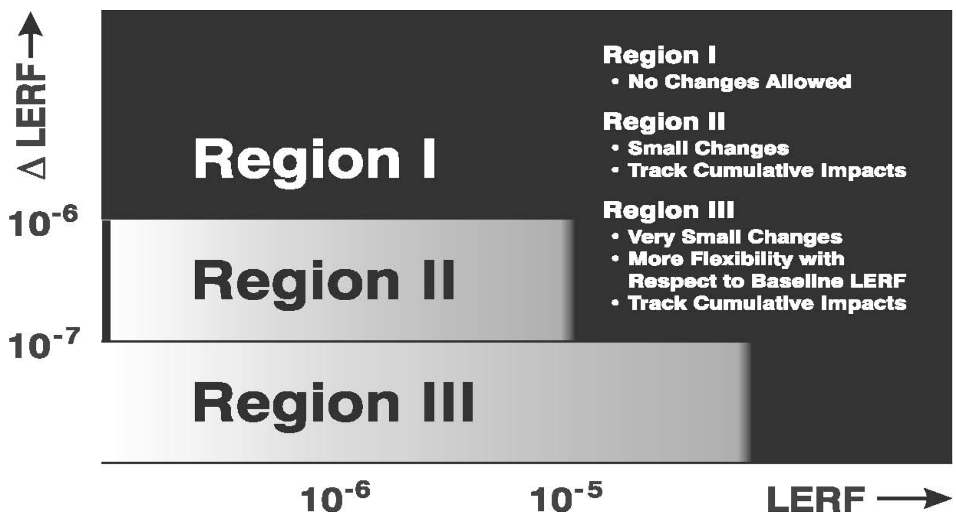

Figure 3. RG 1.174 Acceptance Guidelines for LERF

Although RG 1.174 allows for the use of the Commission's safety goal QHOs in lieu of LERF, it acknowledges that this would require an extension to a Level 3 PRA, and therefore would require additional consideration of the methods, assumptions, and associated uncertainties. Moreover, the acceptance guidelines are intended for comparison with the results of a full-scope PRA that includes all risk contributors. When a limited-scope PRA is used, the contribution of out-of-scope items to risk must be assessed based on the margin between the PRA results and the acceptance guidelines. When the margin is significant, qualitative analyses may be sufficient. When the margin is small, additional PRA analyses may be required.

Importantly, in developing this risk-informed integrated decisionmaking framework, the NRC staff acknowledged that assurance of adequate protection of public health and safety encompasses more than simply demonstrating an acceptable level of overall risk by stating:

"…NRC has chosen a more restrictive policy that would permit only small increases in risk, and then only when it is reasonably assured, among other things, that sufficient defense-in-depth and sufficient margins are maintained. This policy is adopted because of uncertainties and to account for the fact that safety issues continue to emerge regarding design, construction, and operational matters notwithstanding the maturity of the nuclear power industry. These factors suggest that nuclear power reactors should operate routinely only at a prudent margin above adequate protection. The safety goal subsidiary objectives are used as an example of such a prudent margin."

RG 1.174 establishes acceptance guidelines based on CDF and LERF that reasonably assure a prudent margin above adequate protection exists.

Overview of Risk-Informed Regulation in Practice

The NRC now routinely uses risk information to complement traditional deterministic engineering approaches in several components of the NRC's regulatory process, including: licensing and certification, regulations and guidance, oversight, and operational experience. Some of the more significant risk-informed applications involving NPPs within each of these components are highlighted below:

Licensing and Certification
- 10 CFR 52, Licenses, certifications, and approvals for NPPs (includes PRA requirements)

Regulations and Guidance
- 10 CFR 50.44, Combustible gas control for nuclear power reactors
- 10 CFR 50.48(c), Fire protection – National Fire Protection Association Standard 805
- 10 CFR 50.62, Requirements for reduction of risk from anticipated transients without scram (ATWS) events for light-water-cooled NPPs
- 10 CFR 50.63, Loss of all alternating current power (Station blackout rule)
- 10 CFR 50.65, Requirements for monitoring the effectiveness of maintenance at NPPs (Maintenance rule)
- 10 CFR 50.69, Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors

Oversight: Risk-Informed Aspects of the Reactor Oversight Process (ROP)
- Risk-informed baseline inspections
- Risk-informed performance indicators (e.g., Mitigating Systems Performance Index)
- Significance Determination Process (SDP)

Operational Experience: Risk-Informed Programs
- Incident response – Management Directive 8.3[28]
- Event assessment – Accident Sequence Precursor (ASP) Program

[28] Management Directive 8.3, "NRC Incident Investigation Program" (March 27, 2001).